HEALTH PHYSICS

OFFICIAL JOURNAL OF THE HEALTH PHYSICS SOCIETY

Volume 1

KARL Z. MORGAN
Editor-in-Chief

W. S. SNYDER J. A. AUXIER
Editors

OFFICERS AND DIRECTORS OF THE HEALTH PHYSICS SOCIETY

President
LAURISTON S. TAYLOR

Most Recent Past President
FREDERICK P. COWAN

President-Elect
ELDA E. ANDERSON

Secretary
J. W. McCASLIN

Treasurer ROBERT G. GALLAGHAR

Board of Directors
W. D. CLAUS
C. C. GAMERTSFELDER
W. H. LANGHAM
G. V. BEARD
K. Z. MORGAN
J. S. LAUGHLIN
W. E. NOLAN
G. W. C. TAIT
R. M. SIEVERT

MEMBERS OF THE HONORARY EDITORIAL ADVISORY BOARD

Austria: J. J. ZAKOVSKY · *Belgium:* Z. M. BACQ · R. BOULENGER
Canada: F. D. SOWBY · H. E. JOHNS · G. C. BUTLER · *Denmark:* J. C. JACOBSEN · H. LEVI
France: H. P. JAMMET · L. BUGNARD · A. ALLISY · *Germany:* E. H. GRAUL · H. HOLTHUSEN
India: A. S. RAO · *Italy:* A. PERUSSIA · *Japan:* M. K. NAKAIDZUMI · S. SHIMIZU
Netherlands: W. J. OOSTERKAMP · A. H. W. ATEN, JR. · *Norway:* K. KOREN
Sweden: R. M. SIEVERT · L. CARLBOM · *Switzerland:* F. ALDER · G. JOYET · *Turkey:* P. CAMBEL
United Kingdom: W. G. MARLEY · W. BINKS · W. V. MAYNEORD · J. S. MITCHELL · E. E. POCHIN
United States: G. FAILLA · R. D. EVANS · C. L. DUNHAM · A. M. BRUES · P. S. HENSHAW · S. WARREN
J. C. BUGHER · H. A. BLAIR · L. B. SILVERMAN · L. D. MARINELLI · W. T. HAM · B. MOYER
H. M. PARKER · L. SILVERMAN · C. A. TOBIAS · C. R. WILLIAMS · S. T. CANTRIL · S. BLOCK
Uraguay: F. LEBORGNE · *World Health Organization:* I. S. EVE

PERGAMON PRESS NEW YORK LONDON

HEALTH PHYSICS

The Official Journal of the Health Physics Society

EDITORS

DR. K. Z. MORGAN (*Editor-in-chief*), *Director, Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee, U.S.A.*

DR. W. S. SNYDER, *Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee, U.S.A.*

DR. J. A. AUXIER, *Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee, U.S.A.*

INSTRUCTIONS TO CONTRIBUTORS

Manuscripts should be submitted to Dr. K. Z. MORGAN, Health Physics Division, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee. The author's institution is requested to pay a publication charge of $10.00 per page, which entitles the author to 50 free reprints.

Manuscripts submitted for publication in *Health Physics* can be accepted only on the condition that they have not been submitted elsewhere. Manuscripts must be typed, on one side of the paper only, with double or triple spacing and the original and one duplicate copy should be submitted. The figures should be glossy prints of approximately 8 × 10 inches or original line drawings with lettering which will be legible after reduction of the size of the figure. Attach the figures to separate sheets together with the figure number. Captions should be typed on a separate sheet and attached to the manuscript.

An abstract of 200 words or less must accompany the manuscript. It should give a precise account of the major results of the research as well as some indication of methods or applications when these are of interest. Avoid excessively general remarks.

Other considerations of style and form may be in accordance with generally accepted journalistic practices.

Pergamon Press are also publishers of the following journals:

JOURNAL OF NUCLEAR ENERGY (*including* THE SOVIET JOURNAL OF ATOMIC ENERGY*) PART A: REACTOR SCIENCE; PART B: REACTOR TECHNOLOGY
JOURNAL OF INORGANIC AND NUCLEAR CHEMISTRY
TETRAHEDRON (*The International Journal of Organic Chemistry*)
TETRAHEDRON LETTERS
TALANTA (*An International Journal of Analytical Chemistry*)
INTERNATIONAL JOURNAL OF APPLIED RADIATION AND ISOTOPES
BIOCHEMICAL PHARMACOLOGY
* BIOPHYSICS
* JOURNAL OF MICROBIOLOGY, EPIDEMIOLOGY AND IMMUNOBIOLOGY
* PROBLEMS OF HEMATOLOGY AND BLOOD TRANSFUSION
* PROBLEMS OF ONCOLOGY
* PROBLEMS OF VIROLOGY
* SECHENOV PHYSIOLOGICAL JOURNAL OF THE U.S.S.R.
* ELECTRIC TECHNOLOGY, U.S.S.R.
* RADIO ENGINEERING
* RADIO ENGINEERING AND ELECTRONICS
* TELECOMMUNICATIONS
* THE PHYSICS OF METALS AND METALLOGRAPHY
* THE ABSTRACTS JOURNAL OF METALLURGY
* APPLIED MATHEMATICS AND MECHANICS
CHEMICAL ENGINEERING SCIENCE
JOURNAL OF ATMOSPHERIC AND TERRESTRIAL PHYSICS
PLANETARY AND SPACE SCIENCE
GEOCHIMICA ET COSMOCHIMICA ACTA
BULLETIN GÉODÉSIQUE
ANNALS OF THE INTERNATIONAL GEOPHYSICAL YEAR
JOURNAL OF THE MECHANICS AND PHYSICS OF SOLIDS
SPECTROCHIMICA ACTA
ACTA METALLURGICA
(*for the Board of Governors of Acta Metallurgica*)
INTERNATIONAL JOURNAL OF THE PHYSICS AND CHEMISTRY OF SOLIDS
RHEOLOGY ABSTRACTS
DEEP-SEA RESEARCH
JOURNAL OF NEUROCHEMISTRY
JOURNAL OF PSYCHOSOMATIC RESEARCH
JOURNAL OF INSECT PHYSIOLOGY
JOURNAL OF AIR POLLUTION
INTERNATIONAL ABSTRACTS OF BIOLOGICAL SCIENCES
(*for Biological and Medical Abstracts Ltd.*)
VACUUM
OPERATIONAL RESEARCH QUARTERLY
ANNALS OF OCCUPATIONAL HYGIENE
ELECTROCHIMICA ACTA

* *Translations of the Russian Journals published on behalf of the Pergamon Institute, a non-profit-making foundation*

Leaflets giving further details and subscription rates of each of these journals are available on request

LIST OF CONTENTS

NUMBER 1

Foreword 1
L. S. TAYLOR: Brief history of the national committee on radiation protection and measurements (NCRP) covering the period 1929–1946 3
W. BRANDT: Survey of stopping power 11
J. A. AUXIER, G. S. HURST and R. E. ZEDLER: A single ion detector for measurement of γ-ray ionization in cavities 21
R. D. BIRKHOFF, H. H. HUBBELL, JR., J. S. CHEKA and R. H. RITCHIE: Spectral distribution of electron flux in a beta-radioactive medium 27
L. W. COCHRAN: *W* for heavy particles 34
T. E. BORTNER and G. S. HURST: An apparatus for measuring electron attachment: results for oxygen in argon 39
HARALD H. ROSSI and M. LUBERT: Dosimetry with ionization chambers employing internal gas multiplication 46
W. S. SNYDER: Calculation of radiation dose 51
W. D. CLAUS: What is health physics? 56
L. S. TAYLOR: Radiation exposure as a reasonable calculated risk 62
G. H. WHIPPLE: Health physics responsibilities to management 71
F. P. COWAN and J. B. KUPER: Exposure criteria for evaluating the public consequences of catastrophic accidents in large nuclear plants 76
J. E. BALLOU and R. C. THOMPSON: Metabolism of cesium-137 in the rat: Comparison of acute and chronic administration experiments 85
Notes 90
Letters to the Editor 92
News 94
The Second International Course in Health Physics 95
Papers to be published in future issues 96

NUMBER 2

LAURISTON S. TAYLOR: History of the International Commission on Radiological Protection (ICRP) . . . 97
WRIGHT H. LANGHAM: Potential hazard of world-wide Sr^{90} fallout from nuclear weapons testing . . . 105
K. Z. MORGAN: Current status of the internal dose problem 125
P. R. BOLTON, E. C. BARNES, E. D. DURKOSH and W. D. SMALL: Aerosol activity from experimental low power reactor operation 135
PAUL S. HENSHAW: Assessing the impact of environmental radiation on total health 141
K. E. COWSER and F. L. PARKER: Soil disposal of radioactive wastes at ORNL: Criteria and techniques of site selection and monitoring 152
G. W. C. TAIT and W. F. MERRITT: Emergency radiation monitoring of drinking water 164
P. W. REINHARDT and F. J. DAVIS: Improvements in the threshold detector method of fast neutron dosimetry . 169
S. G. PEARSALL, L. GEMMELL and A. BRESLIN: Possible health hazards associated with the laundering of radioactively contaminated protective clothing 176
P. J. BARRY: Some general considerations in chemical decontamination 184
A. BRODSKY, L. W. FAGG and T. D. HANSCOME: A short procedure for electrodepositing fissionable elements . 189
Notes
J. R. MANN: Converting a gas flow alpha proportional counter to the scintillation type 192
R. L. KOONTZ and C. T. NELSON: A continuous automatic environmental air sampler 194
WALLACE DE LAGUNA: Desert basins and waste disposal 195
LAURISTON S. TAYLOR: Maximum permissible radiation exposures to man 200
Letter to the Editor
J. J. FITZGERALD: Suggested insignias for the Health Physics Society 205
News 207
Papers to be published in future issues 212
Abstracts and titles of papers presented at the Third Annual Meeting of the Health Physics Society . . . 215
The Second United Nations International Conference on the Peaceful Uses of Atomic Energy. A selection of abstracts and titles of papers of interest to Health Physicists 235

NUMBER 3

G. M. DUNNING: Radiation exposures from nuclear tests at the Nevada Test Site 255
R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD: The naturally occurring α-ray activity of foods . . 268
B. R. FISH: Applications of an analog computer to analysis of distribution and excretion data . . . 276
C. W. MAYS, D. H. TAYSUM, W. FISHER and B. W. GLAD: Bremsstrahlung counting of Sr^{90} injected dogs . 282
S. R. BERNARD: Maximum permissible amounts of natural uranium in the body, air and drinking water based on human experimental data 288
LAURISTON S. TAYLOR: History of the International Commission on Radiological Units and Measurements (ICRU) 306
J. NEUFELD: On the relationship between the charge of an ion and its velocity 315
E. TOCHILIN and G. D. KOHLER: Neutron beam characteristics from the University of California 60 in. cyclotron 332
W. C. ROESCH, R. C. MCCALL and F. L. RISING: A pulse reading method of condenser ion chambers . . 340
I. C. NELSON: Electronic data processing and radiation exposure records 345
Notes
L. SCHWENDIMAN: Probability of human contact and inhalation of particles 352
G. E. WILCOX: A projection microscope for neutron dosimetry 355
Letter to the Editor
S. BLOCK: Some comments on the decay scheme and dosimetry of Cs^{137} 357
News 358
Papers to be published in future issues 362
Errata 362

NUMBER 4

DIXON CALLIHAN and JOSEPH T. THOMAS: Accidental radiation excursion at the Oak Ridge Y-12 Plant—I. Description and physics of the accident 363
BETSY J. STOVER: Metabolism of radioactive isotopes which deposit mainly in the skeleton 373
G. M. DUNNING: External β-doses from radioactive fallout 379
R. H. RITCHIE and G. S. HURST: Penetration of weapons radiation: application to the Hiroshima–Nagasaki studies 390
E. L. GEIGER: Radioassay of uranium and plutonium in vegetation, soil and water 405
A. WALTON, R. KOLOGRIVOV and J. L. KULP: The concentration and distribution of radium in the normal human skeleton 409
H. P. YOCKEY: Radiation aging and its relation to the principles of Health Physics 417
ANN T. NELMS and J. W. COOPER: U^{235} fission product decay spectra at various times after fission . . . 427
C. F. MACGOWAN: Comments on the labor relations aspect of nuclear energy 442
Note
C. T. NELSON: A portable calibrator for β-γ survey instruments using Sr^{90} 447
Letter to the Editor
J. V. NEHEMIAS: Suggested insignia for the Health Physics Society 449
News 450
Papers to be published in future issues 454
Erratum 454

AUTHOR INDEX

AUXIER, J. A. p. 21
BALLOU, J. E. p. 85
BARNES, E. C. p. 135
BARRY, P. J. p. 184
BERNARD, S. R. p. 288
BIRKHOFF, R. D. p. 27
BLOCK, S. p. 357
BOLTON, P. R. p. 135
BORTNER, T. E. p. 39
BRANDT, W. p. 11
BRESLIN, A. p. 176
BRODSKY, A. p. 189
CALLIHAN, DIXON p. 363
CHEKA, J. S. p. 27
CLAUS, W. D. p. 56
COCHRAN, L. W. p. 34
COOPER, J. W. p. 427
COWAN, F. P. p. 76
COWSER, K. E. p. 152
DAVIS, F. J. p. 169
DUNNING, G. M. pp. 255, 379
DURKOSH, E. D. p. 135
FAGG, L. W. p. 189
FISH, B. R. p. 276
FISHER, W. p. 282
FITZGERALD, J. J. p. 205
GEIGER, E. L. p. 405
GEMMELL, L. p. 176
GLAD, B. W. p. 282
HANSCOME, T. D. p. 189
HENSHAW, PAUL S. p. 141
HUBBELL, JR., H. H. p. 27
HURST, G. S. pp. 21, 39, 390
KOHLER, G. D. p. 332
KOLOGRIVOV, R. p. 409
KOONTZ, R. L. p. 194
KULP, J. L. p. 409
KUPER, J. B. p. 76
LAGUNA, WALLACE DE p. 195
LANGHAM, WRIGHT H. p. 105
LUBERT, M. p. 46
MACGOWAN, C. F. p. 442
MANN, J. R. p. 192
MAYNEORD, W. V. p. 268
MAYS, C. W. p. 282
MCCALL, R. C. p. 340
MERRITT, W. F. p. 164
MORGAN, K. Z. p. 125
NEHEMIAS, J. V. p. 449
NELMS, ANN T. p. 427
NELSON, C. T. pp. 194, 447
NELSON, I. C. p. 345
NEUFELD, J. p. 315
PARKER, F. L. p. 152
PEARSALL, S. G. p. 176
RADLEY, J. M. p. 268
REINHARDT, P. W. p. 169
RISING, F. L. p. 340
RITCHIE, R. H. pp. 27, 390
ROESCH, W. C. p. 340
ROSSI, HARALD H. p. 46
SCHWENDIMAN, L. p. 352
SMALL, W. D. p. 135
SNYDER, W. S. p. 51
STOVER, BETSY J. p. 373
TAIT, G. W. C. p. 164
TAYLOR, LAURISTON S. pp. 3, 62, 97, 200, 306
TAYSUM, D. H. p. 282
THOMAS, JOSEPH T. p. 363
THOMPSON, R. C. p. 85
TOCHILIN, E. p. 332
TURNER, R. C. p. 268
WALTON, A. p. 409
WHIPPLE, G. H. p. 71
WILCOX, G. E. p. 355
YOCKEY, H. P. p. 417
ZEDLER, R. E. p. 21

FOREWORD

THIS, the first issue of *Health Physics*, the journal of the Health Physics Society, marks the formal appearance of the field of Health Physics in the family of scientific disciplines. Like the Health Physics Society, this journal is intended to be international in scope and interest. In this nuclear age, hazards of ionizing radiation are increasing in giant steps, measured in orders of magnitude, and they are basically the same irrespective of the countries in which they occur. The genetic problems associated with occupational exposure to ionizing radiation are of international significance, and the river which flows from one country to the next, picking up radioactive contaminants along its course and discharging them into the area, recognises no national standards of radiation protection. Thus, no single national organization or national journal can truly represent or adequately serve the profession of Health Physics.

Some of the hazards of ionizing radiation were recognized only a few weeks after the discovery of X-rays and though we have been learning more about these hazards ever since, we still have far to go before we have a complete understanding of the mechanisms of radiation damage and of the full implications and consequences of somatic and genetic damage, either directly to man or indirectly through the ecology of his environment.

Radiologists and medical physicists, with the help and guidance of the International Commission on Radiological Protection and of the various national committees, have done a remarkable job of minimizing radiation hazards. With the exception of the unfortunate experience of the radium dial painting industry and of the early quasi-medical use of radium and X-rays, there have been very few cases of serious over-exposure.

On December 2, 1942, the first nuclear reactor was operated at the University of Chicago, and at that time it was recognized that behind the concrete shield of a reactor is the equivalent of many thousand times the radiation from all the radium available in all the world. It was then that health physics had its start—at first only six health physicists at the University of Chicago, while today it is estimated there are over 2000 practicing health physicists in all parts of the world.

In June 1955 at the Ohio State University, the health physicists first organized, and in June 1956 at the University of Michigan the constitution and by-laws were adopted and the name "Health Physics Society" was selected. By the time the Society met in June 1957 at the University of Pittsburgh it had grown in membership to about 900, including an encouraging representation from many countries of the world. Those attending this meeting voted unanimously to publish a journal. It was agreed that rather than publish proceedings as was done following the meetings in Ohio and Michigan, a selection of papers from the Pittsburgh meeting would be published in the

1

first few issues of the journal. This has been an invaluable source of material for the launching of the journal.

It will be the policy of this journal to accept for publication only those articles which deal with or relate to some aspect of Health Physics, which in turn is defined as a study and practice dealing with any and all factors relating to damage from ionizing radiation and the prevention of such damage. This may mean considerable overlap into the fields of physics, chemistry, biology, medicine, geology, etc., but in every case our objective will be to localize and concentrate in one journal papers relating to radiation protection. These papers will be of four general categories: (1) Research, (2) Engineering, (3) Applied and (4) General. The research papers will include original contributions from both theoretical and experimental research and will relate to such subjects as mechanisms of radiation damage, a determination of the parameters used in the calculation of dose or in the design of a dosimeter, radiobiological studies, ecological effects of radiation, etc. Engineering papers will relate to applied research such as radioactive waste disposal, instrument development, laundry decontamination studies, etc. Applied health physics papers will cover a wide range of interest and experience and will include such subjects as personnel monitoring, area monitoring, use of instruments, urine analysis for radioactive materials, radiation accidents, education and training in health physics, fall-out studies, etc. The "general" category is for a variety of papers on radiation protection; papers that do not fit into the first three classifications. For example, this might include papers on the history and philosophy of the national and international organizations for radiation protection, changes in maximum permissible levels of exposure, liability insurance against radiation hazards, etc. Survey papers which provide timely and authoritative summaries of areas of general interest in any of these categories will be welcomed. In addition to the regular papers, there will be a section of "notes" where short and/or preliminary communications will be published and where brief news items of general interest will appear.

We believe that the nuclear age is here to stay and that its future rests in large measure on the successful control of radiation exposure. *Health Physics* will attempt to uphold the high professional standards of the Health Physics Society and will do its utmost to disseminate knowledge in this field. A successful nuclear industry perhaps entails an increased exposure of mankind to ionizing radiation. We must understand the full and ultimate consequences of this exposure and limit it at a level where we, and those that come after us, can reap the maximum benefits of this new age. May the pages of this journal help us and our associates to understand how much radiation exposure is permissible and guide us in the best means of measuring and controlling it to the greatest good of mankind.

K. Z. Morgan

Health Physics Pergamon Press 1958. Vol. 1. pp. 3–10

BRIEF HISTORY OF THE NATIONAL COMMITTEE ON RADIATION PROTECTION AND MEASUREMENTS (NCRP) COVERING THE PERIOD 1929–1946

LAURISTON S. TAYLOR
Chairman National Committee on Radiation Protection and Measurements

(*Received* 12 *December* 1957)

Abstract—The National Committee on Radiation Protection and Measurements was established in the United States in 1929, since which time it has provided the basic standards and guidance in the field. Through the co-operation of many organizations this has proven to be an effective program. At the same time through interlocking membership between the NCRP and the International Commission on Radiation Protection, the views of this country have been integrated into the world views. A brief listing of the activities of the NCRP and its reports is given.

1. FORMATION

THE roots of the National Committee on Radiation Protection and Measurements go back to 1928 and are intimately related to the formation of the International Commission on Radiological Protection in July of that year. With the possibility in mind of forming an international organization on radiological protection, the Second International Congress of Radiology, before meeting in Stockholm in July 1928, invited several countries to send representatives to the Congress for the purpose of discussing protection problems and possibly preparing some initial X-ray protection recommendations. From the United States, L. S. TAYLOR was designated as representative of the National Bureau of Standards, and one representative each attended for the American Roentgen Ray Society and the Radiological Society of North America.

When attempts were made to reach agreement between the United States and other countries, serious difficulties arose. Each of our two radiological societies offered different recommendations and each claimed to be the authoritative body. The NBS had no recommendations to offer and was there more by way of an observer. As a result, the recommendations that were in fairly acceptable form, prepared by the British protection committee, were adopted as the first international recommendations. In the process, the United States delegates showed up rather poorly, in that agreement could not be reached on who authoritatively represented the views of the United States.

Germany presented a somewhat similar though not quite so serious a situation as had the U.S., in that its representatives at the preliminary discussions also could not agree on who carried the necessary authority.

Concurrent with the meetings of the Congress, G. W. C. KAYE and STANLEY MELVILLE (Great Britain) and L. S. TAYLOR (United States) set about to organize a permanent structure for an international organization. After preliminary discussions, during which some general rules of organization were developed, the International Commission on Radiological Protection* was organized, the membership consisting of the above-mentioned

* Until 1946, the Commission was called the International X-ray and Radium Protection Committee.

persons, Dr. ROLF SIEVERT of Sweden and Dr. ISER SOLOMON of France and Dr. GUSTAV GROSSMAN of Germany. It was agreed by this group that the Commission should be kept small, and that wherever possible, representatives to the Commission should be chosen from National Laboratories where such laboratories existed in member countries. This arrangement and the general philosophy of operation of the Commission was approved by the Second International Congress of Radiology before the close of its sessions.

Because of the confusion regarding accredited representation, introduced primarily by the United States but also by Germany and to a lesser extent by France, the chairman of the ICRP, Dr. G. W. C. KAYE, recommended that a single central committee be established within those countries having more than one radiological organization for the purpose of consolidating national recommendations for presentation at the next meeting of the Commission. It was suggested that the members of the ICRP take this up individually with the various groups in their countries.

As the U.S. representative to the international protection group, it became the responsibility of L. S. TAYLOR to convey its recommendation to the various groups involved in this country, to convince them of its soundness, to obtain their approval and suggestions, and to organize a national committee which could deal most effectively with the protection problems faced at that time.

In September 1928, this question was discussed informally with the president of the American Roentgen Ray Society, at its annual meeting in West Baden, Indiana. Similar discussions were held with the president of the Radiological Society of North America in December of that year. As a result of these discussions, these organizations agreed to consolidate their protection activities into a single committee. They further recommended that the committee's activities be centralized at the National Bureau of Standards for the following reasons:

(1) It had by that time established a definite long-range program in the general field of radiation protection.

(2) It had the only laboratory in the country having as its primary interest the development of radiation-protection data and information.

(3) It had no inter-society or political ties and therefore could be expected to retain an independent position and viewpoint.

(4) It provided the official U.S. representative to the International Commission on Radiological Protection.

The two radiological societies each recommended a physicist and a radiologist for membership in the proposed national committee and the American Medical Association appointed a member to represent its viewpoints. It was also felt that representation of the X-ray equipment manufacturers would be desirable and each of the manufacturers was asked to nominate candidates for this representation. Of the nominations received, the manufacturers then chose two to serve as their representatives.

Thus, early in 1929, the initial organization of the "Advisory Committee on X-ray and Radium Protection" was established with L. S. TAYLOR acting as chairman and with the following participating organizations and representatives:

American Roentgen Ray Society:
H. K. PANCOAST and J. L. WEATHERWAX

Radiological Society of North America:
R. R. NEWELL and G. FAILLA

American Medical Association:
F. CARTER WOOD

X-ray Equipment Manufacturers:
W. D. COOLIDGE and W. S. WERNER

National Bureau of Standards and ICRP:
L. S. TAYLOR

2. HISTORY

The first meeting of the committee was held in September 1929, during the annual meeting of the American Roentgen Ray Society. As its first objective, the committee undertook the preparation of recommendations on X-ray protection. These were published on 16 May 1931, as National Bureau of Standards Handbook 15.

The next effort was directed towards the preparation of recommendations on radium protections. For this purpose, Dr. L. F. CURTISS was named to the committee as the

NBS representative for radium protection recommendations, and Dr. C. F. BURNAM as the representative of the American Radium Society. The first handbook on radium protection, NBS Handbook 18, was prepared by Drs. CURTISS, BURNAM, FAILLA, NEWELL, WEATHERWAX and WOOD, and was published 17 March 1934.

Soon after the publication of Handbook 15 on X-ray Protection, very rapid developments were made in the X-ray field; by 1934 or 1935 it was recognized that this handbook would have to be revised. This task was undertaken by the original committee, except for the replacement of Dr. PANCOAST by Dr. EUGENE P. PENDERGRASS as representative of the American Roentgen Ray Society. The revised recommendations were issued in July, 1936 as NBS Handbook 20.

It might be worth noting that in this handbook, there appeared for the first time the recommendation of a specific permissible exposure level (then called tolerance dose) of radiation that could be allowed for occupational exposure. The figure recommended was 0.1 r/week. This permissible exposure level remained in force for 12 years and was used by the Manhattan District in its operations. It was subsequently changed as a result of NCRP action in about 1948.

The revision of Handbook 18 on radium protection was next undertaken and the new handbook (H 23) was issued 25 August 1938.

These two handbooks, H 20 and H 23, were accepted in this country as the primary guides for protection against X-rays and the radiations from radium. As noted above, they were also the primary guides in this field to the Manhattan project.

Through the war years, there was no formal activity by the Advisory Committee. During that time, however, most of the members of the Advisory Committee were drawn into the Manhattan District program and it was largely through their efforts that uniform safety regulations prevailed during that period.

During its early activities, it was customary for the full committee to work together on the development of protection recommendations. When completed, the recommendations were submitted through their respective representatives to the participating organizations for noting and approval. Formal approval was usually given at one of the regular business meetings of the societies. With the NBS as one sponsor of the committee, the recommendations were published by the Government Printing Office as National Bureau of Standards Handbooks thus receiving the usual NBS editorial processing.

In September 1946, an informal meeting of the Advisory Committee was held to discuss the extensive revision needed in the X-ray protection recommendations, particularly in the upper voltage regions. At this meeting, it was pointed out that protection problems had become too complex to permit their study and solution by the committee as then constituted. It was recommended that steps be taken to secure the participation in this work of additional groups such as the Manhattan District and U.S. Public Health Service, Military Department, etc. This recommendation was presented to Dr. E. U. CONDON, then Director of NBS, who communicated with the Manhattan District and the Public Health Service on 6 October 1946, inviting their participation through appointment of two representatives each (one physicist and one radiologist). In response to this invitation, In October 1946, Dr. S. L. WARREN and Dr. K. Z. MORGAN were appointed as representatives of the Manhattan District, and the Public Health Service named Dr. H. L. ANDREWS and Dr. E. G. WILLIAMS.

The first normal post-war meeting of the Committee was held on 4 December 1946. In the agenda for this meeting, it was pointed out that new data had become available since the issuance of the recommendations on X-ray protection, and that many new protection problems had arisen with the rapid expansion in the radiation field (protection against neutrons, multi-million volt X-rays, radioactive isotopes, etc.). It was suggested that the scope of the work be defined; and that consideration be given to organizing small working groups to deal with each of the problems, their completed reports to be submitted to the committee for approval.

3. ORGANIZATION

Discussions along these lines were held at the 4 December 1946 meeting, and as a result, it was agreed that the Committee should be substantially enlarged and reorganized. At the same time, it was felt that the name of the Committee should be made more inclusive and it was therefore renamed National Committee on Radiation Protection. The National Bureau of Standards was reaffirmed as the central coordinating agency for the work of the Committee; sponsorship by an impartial agency was felt to be particularly advantageous in view of the various types of participating organizations (radiological societies, industry, government, and the possible inclusion of industrial and labor groups).

The general organization and operational procedures outlined below were agreed upon at this meeting, and have been the basis for the continuing operation of the Committee:

(1) The committee would consist of an Executive Committee, Main Committee, and as many subcommittees as necessary to consider the problems that come within the Committee's scope.

(2) The Executive Committee would be composed of five members appointed by the chairman and subject to the approval of the Main Committee. The Committee chairman would act as chairman of the Executive Committee.

(3) The Main Committee would be composed of (a) technically qualified representatives appointed by organizations interested in the scientific and technical aspects of radiation protection, (b) representatives at large whose services are felt to be of special value, appointed by the Executive Committee, and (c) chairmen of subcommittees.

(4) The choice of chairmen and members of subcommittees would not be restricted to members of the Main Committee but would be based on the particular qualifications needed for the work. (In organizing subcommittee memberships, the following practice has been and still is followed: The subcommittee chairman is selected by the Committee chairman with the approval of the Executive Committee; the subcommittee chairman chooses his working group, makes informal contacts, and submits to the Committee chairman his membership selections; the Committee chairman issues formal membership invitations to serve on the subcommittees. Additions may be made to the subcommittee membership if particular specialized information is found to be needed, or individuals may be invited to serve as consultants to the group with due acknowledgement of their assistance included in the published recommendations.)

(5) The final reports of the subcommittee would be submitted to the Executive and Main Committees for approval. Because of the high degree of success of the NBS Handbook series, it was recommended that this mode of publication and distribution to the public be continued.

Because of the reorganization and enlargement of the Committee, the chairmanship was thrown open for reconsideration. L. S. TAYLOR was nominated and approved by vote to continue indefinitely in this capacity.

Discussions were held regarding the organizations that might appropriately be invited to participate and the suggestions made were used as a basis for the subsequent enlargement of the representation on the Main Committee. It was agreed to establish the following subcommittees:

(1) Permissible external dose.

(2) Permissible internal dose.

(3) X-rays up to 2 MeV.

(4) Heavy ionizing particles (neutrons, protons and heavier).

(5) Electrons, radium, and X-rays above 2 MeV.

(6) Radioactive isotopes, fission products, including their handling and disposal.

(7) Monitoring methods and instruments.

With the formulation of these basic philosophies, the Committee began its active program. Its accomplishments and growth since its reorganization in 1946 can be seen by the appended list of handbooks published to date, and the appended membership list showing the present representation, subcommittee structure, and complete membership.

MEMBERSHIP LIST
NATIONAL COMMITTEE ON RADIATION PROTECTION AND MEASUREMENT (NCRP)

Chairman:

L. S. Taylor

Secretary

Sarah W. Raskin

Executive Committee

Members

L. S. Taylor, Chairman
E. C. Barnes
C. B. Braestrup
C. L. Dunham
H. B. Glass
H. M. Parker
C. Powell
R. S. Stone
S. Warren

Consultants

J. C. Bugher
G. Failla
E. G. Williams

Main Committee (and organization represented)

H. L. Andrews, USPHS and Subcommittee Chairman
E. C. Barnes, Amer. Indust. Hygiene Assn.
A. C. Blackman, Internl. Assn. of Govt. Labor Officials
C. B. Braestrup, Radiol. Soc. of North America and Subcommittee Chairman
J. C. Bugher, Representative at large
R. H. Chamberlain, Amer. College of Radiology
W. D. Claus, USAEC
C. L. Dunham, USAEC
T. P. Eberhard, Amer. Radium Society and Subcommittee Chairman
T. C. Evans, Amer. Roentgen Ray Society
G. Failla, Representative at large
J. W. Healy, Health Physics Society and Subcommittee Chairman
P. C. Hodges, Amer. Medical Assn.
H. W. Koch, Subcommittee Chairman
W. Langham, Subcommittee Chairman
R. M. Lechausse, Col., U.S. Air Force
W. B. Mann, Subcommittee Chairman
G. M. McDonnell, Lt. Col., U.S. Army
G. W. Morgan, Subcommittee Chairman
K. Z. Morgan, Health Physics Society and Subcommittee Chairman
R. J. Nelsen, Amer. Dental Assn.
R. R. Newell, Amer. Roentgen Ray Society
C. Powell, USPHS
E. H. Quimby, Amer. Radium Society and Subcommittee Chairman
S. W. Raskin, NBS
J. A. Reynolds, Natl. Electrical Mfgr. Assn.
H. H. Rossi, Subcommittee Chairman
M. D. Schulz, Amer. College of Radiol.
L. S. Skaggs, Subcommittee Chairman
J. H. Sterner, Amer. Indust. Hygiene Assn.
R. S. Stone, Radiol. Soc. of North America
I. R. Tabershaw, Internl. Assoc. of Govt. Labor Officials
L. S. Taylor, NBS
E. D. Trout, Natl. Electrical Mfgr. Assn.
S. Warren, Representative at large
J. L. Weatherwax, Representative at large
E. G. Williams, Representative at large
S. F. Williams, Capt., U.S. Navy
H. O. Wyckoff, Subcommittee Chairman

Subcommittee 1

Permissible dose from external sources

A. H. Dowdy
G. Failla
H. Friedell
H. J. Muller
H. M. Parker
C. Stern
R. S. Stone

Note: the work of this subcommittee is temporarily the responsibility of the Executive Committee together with the subcommittee members.

Subcommittee 2

Permissible internal dose

K. Z. Morgan, Chairman
A. M. Brues
P. Durbin
G. Failla
J. Healy
L. D. Marinelli
J. E. Rose
Shields Warren

Subcommittee 3

X-rays up to two million volts

T. P. Eberhard, Chairman
C. B. Braestrup
E. Focht
J. Hale

E. Miller
J. M. Mozley
R. J. Nelsen
R. R. Newell
S. W. Smith
E. D. Trout

Subcommittee 4

Heavy particles (neutrons, protons, and heavier)

H. H. Rossi, Chairman
E. P. Blizard
R. S. Caswell
F. P. Cowan
D. B. Cowie
T. C. Evans
D. J. Hughes
L. D. Marinelli
W. S. Snyder
C. A. Tobias

Subcommittee 5

Electrons, gamma rays and X-rays above two million volts

H. W. Koch, Chairman
G. C. Baldwin
C. B. Braestrup
D. B. Cowie
U. Fano
J. S. Laughlin
L. D. Marinelli
D. Scag
L. S. Skaggs

Subcommittee 6

Handling of radioactive isotopes and fission products

J. W. Healy, Chairman
P. C. Aebersold
G. Failla
S. Feitelberg
D. Hull
L. D. Marinelli
H. M. Parker
J. E. Rose
W. K. Sinclair
M. M. D. Williams

Subcommittee 7

Monitoring methods and instruments

H. L. Andrews, Chairman
C. B. Braestrup
J. Healy
R. Lapp
W. H. Ray
J. E. Rose
E. G. Williams

Subcommittee 8

Waste disposal and decontamination

Note: This subcommittee has been inactivated.

Subcommittee 9

Protection against radiations from Ra, Co^{60}, *and* Cs^{137} *encapsulated sources*

C. B. Braestrup, Chairman
H. Blatz
M. Brucer
T. P. Eberhard
G. Ferlazzo
E. H. Quimby
E. L. Saenger
W. K. Sinclair
H. O. Wyckoff

Subcommittee 10

Regulation of radiation exposure dose

W. A. McAdams, Chairman
W. E. Chamberlain
P. C. Hodges
R. R. Newell
E. P. Pendergrass
L. S. Taylor
J. G. Terrill, Jr.
E. D. Trout
F. Western
E. G. Williams

Subcommittee 11

Incineration of radioactive waste

G. W. Morgan, Chairman
R. C. Corey
S. Feitelberg
W. H. Langham
J. A. Lieberman
L. Silverman

Subcommittee 12

Electron protection

L. S. Skaggs, Chairman
E. A. Burrill
H. W. Koch

J. S. Laughlin
R. F. Post
E. D. Trout

Subcommittee 13

Safe handling of cadavers containing radioactive isotopes

E. H. Quimby, Chairman
S. Feitelberg
L. R. Peet
W. B. Stewart
R. Yalow
E. G. Maier, Consultant

Subcommittee 14

Permissible exposure doses under emergency conditions

Members

L. S. Taylor, Acting Chairman
H. L. Andrews
H. Blair
G. Casarett
G. Dunning
E. Green
P. Henshaw
R. Hasterlik
G. V. Leroy
E. G. Williams

Consultants

C. B. Braestrup
R. F. Brown
J. Bugher
A. H. Dowdy
L. H. Garland
L. H. Hempelmann
R. D. Huntoon
H. M. Parker
G. A. Sacher
R. S. Stone
Shields Warren

Subcommittee M-1

Standards and measurement of radio-activity for radiological use

Members

W. B. Mann, Chairman
T. P. Eberhard
K. W. Geiger
W. Gross
R. Rugh
W. K. Sinclair

Consultants

P. J. Campion
C. L. Comar
S. D. Garfinkle
E. H. Quimby
S. A. Reynolds
H. H. Seliger

Subcommittee M-2

Standards and measurement of radiological exposure dose

Members

H. O. Wyckoff, Chairman
C. B. Braestrup
R. S. Caswell
C. Garrett
J. Hale
J. S. Laughlin
R. R. Newell

Consultants

W. A. Jennings
A. McCrea
G. R. Ringo
R. H. Ritchie
R. Robbins
E. Tochilin

Subcommittee M-3

Standards and measurement of absorbed radiation dose

Members

H. O. Wyckoff, Chairman
G. S. Hurst
H. W. Koch
H. M. Parker
W. C. Roesch
H. H. Rossi
G. N. Whyte

Consultants

F. H. Attix
R. S. Caswell
D. V. Cormack
W. Gross
H. E. Johns
F. C. Maienschein
J. W. Motz
J. A. Sayeg
R. H. Schuler
R. W. Wallace

Subcommittee M-4

Relative biological effectiveness

Members

W. H. Langham, Chairman
V. P. Bond
R. D. Evans
T. C. Evans
M. P. Finkel
J. N. Stannard
J. B. Storer
C. A. Tobias
A. C. Upton

Consultants

H. A. Blair
A. M. Brues
H. I. Kohn
R. E. Zirkle

Recommendations of NCRP 1931–1957

NBS Handbook No.	Title	Date
15	X-ray protection Superseded by H20	16 May 1931
18	Radium protection for amounts up to 300 mg Superseded by H23	17 March 1934
20	X-ray protection Superseded by H41	24 July 1936
23	Radium protection Superseded by H54	25 Aug. 1938
41	Medical X-ray protection up to two million volts Superseded by H60	30 March 1949
42	Safe handling of radioactive isotopes	Sept. 1949
48	Control and removal of radioactive contamination in laboratories	15 Dec. 1951
49	Recommendations for waste disposal of phosphorus-32 and iodine-131 for medical users	2 Nov. 1951
51	Radiological monitoring methods and instruments	7 April 1952
52	Maximum permissible amounts of radioisotopes in the human body and maximum permissible concentrations in air and water.	20 March 1953
53	Recommendations for the disposal of carbon-14 wastes	26 Oct. 1953
54	Protection against radiations from radium, cobalt-60, and cesium-137	1 Sept. 1954
55	Protection against betatron-synchrotron radiations up to 100 million electron volts	26 Feb. 1954
56	Safe handling of cadavers containing radioactive isotopes	26 Oct. 1953
58	Radioactive-waste disposal in the ocean	25 Aug. 1954
59	Permissible dose from external sources of ionizing radiation	24 Sept. 1954
60	X-ray protection	1 Dec. 1955
61	Regulation of radiation exposure by legislative means	9 Dec. 1955
63	Protection against neutron radiation up to 30 million electron volts	22 Nov. 1957

Health Physics Pergamon Press 1958. Vol. 1. pp. 11–20

SURVEY OF STOPPING POWER*

WERNER BRANDT
Du Pont Radiation Physics Laboratory, Wilmington, Del.

(*Received* 20 *January* 1958)

Abstract—This survey discusses the status of stopping power problems pertinent to dosimetry. It is shown that stopping powers of light elements can be predicted with high accuracy over wide energy ranges. A systematic discrepancy of ca. 2 per cent still exists between the stopping powers of heavy elements derived from measurements with high-energy particles, and those derived from measurements with low-energy particles. This discrepancy may arise from an underestimate of the stopping-power deficiency of inner atomic shells at low energies as introduced by the presently available theoretical inner-shell corrections. Only the Z-dependence of mean excitation potentials derived from high-energy experiments is in concord with the statistical Thomas–Fermi–Dirac model of the atom. The non-additivity of stopping powers can be incorporated nearly quantitatively in stopping power calculations by applying valence-state corrections to the mean excitation potentials of atoms.

1. INTRODUCTION

CONSIDER a beam of charged particles impinging on a target of thickness Δx with a kinetic energy E_1. If Δx is sufficiently small, the particles will penetrate the target and emerge with an energy $E_2 < E_1$. The average energy loss per unit distance traversed in the target defines the *stopping power* of the target material with respect to the kind and energy of the penetrating particles:

$$-\frac{dE}{dx} = \left(\frac{\bar{E}_2 - E_1}{\Delta x}\right)_{\Delta x \to 0}. \qquad (1)$$

The numerically identical expression is termed *linear energy transfer* (LET) provided the radiation losses are small, and when attention is focused on the energy transferred to the irradiated material rather than on the slowing-down process of the particles. The emerging beam, if mono-energetic and well collimated before entering the target, will have an energy distribution about the mean $\bar{E}_2$ and a spatial distribution, signifying the statistical nature of the stopping processes on the atomic scale.

The stopping of charged particles is caused by the Coulomb interaction between the moving particles and the atoms in the stopping substance. The objective, then, of the study of stopping powers is a general description of the energy transfer from the moving particles to the target in terms of the basic properties of both the particles and the stopping atoms. Once such a description has been achieved and proven to be correct, stopping experiments can serve either to characterize new high-energy particles by their energy loss in known targets, or to uncover with known particles the modes of energy transfer in new target materials.

The phenomenon of stopping, discovered by LEITHÄUSEN[1] in 1904, has influenced profoundly the development of the atomic model. BOHR[2] formulated the first atomic theory of stopping in his classic papers of 1913 and 1915. The theory was developed fully within the framework of quantum mechanics by BETHE[3] in 1930, actually long before a quantitative experimental confirmation was possible. BLOCH[4] applied the statistical Thomas–Fermi model of the atom to BETHE's theory with the result that the stopping power should be a simple function of the atomic number of the stopping atoms. In 1939 and 1940, FERMI[5] developed the theory of the effect of the polarization on the stopping power of a dielectric with a single dispersion frequency, a theory subsequently to be extended by several investigators to multi-dispersive systems.[6–8]

* Presented at the Symposium on Physical Principles of Dosimetry, held by the Health Physics Society, on 17 June, 1957, in Pittsburgh, Pennsylvania.

During the last ten years, the study of stopping powers has attracted widespread interest, mainly for two reasons. First, particles of sufficiently high energies have become available in the laboratory to subject the theory of stopping to quantitative experimental tests. These tests show an amazingly detailed agreement between theory and experiment. Still, as will be discussed shortly, some discrepancies need further clarification, which can be achieved only by extremely accurate experiments. Secondly, the rapid progress in radiation research has brought to light the role of stopping powers in chemical and biological irradiation effects, and the many problems connected with their characterization.

Dosimetry, in particular, depends significantly on the precision of stopping-power data. The Bragg–Gray cavity principle of dosimetry states that the ratio of the energies absorbed in small but equal volumes of a dense medium and of a gas is equal to the ratio of the respective stopping powers. Since the energy absorbed in the gas can be measured with high precision as an ionization current, the accuracy of dosimetry in condensed systems is often limited by the accuracy of experimental or calculated relative stopping powers.

2. SCOPE OF SURVEY

Several excellent reviews of stopping powers have appeared recently.[9–12] Therefore, this survey gives only a cursory outline of the main steps of reasoning which lead to a theoretical expression for the stopping power of a substance. It then concentrates on two subjects of current interest, viz. the velocity dependence of mean excitation potentials and the additivity of stopping powers. This discussion emphasizes the effects of various atomic factors on stopping, but without embarking on a detailed theoretical discourse.

3. THEORY OF STOPPING

Suppose a charged particle interacts with an electron virtually at rest in the target. If the interaction is considered as an inelastic collision, energy is transferred to the electron, exciting it to higher states. One can calculate the average energy transferred during the encounter by obtaining, in terms of quantum mechanics, the matrix elements for all the transitions of the electron induced by the field of the passing particle, and averaging over all initial and final states. The stopping power of the medium is then simply the sum of the contributions from all its electrons.

The calculation makes use of BORN's approximation, which requires that the orbital "velocity" of the electrons is small compared to the velocity of the particle. For all but the lightest elements, this condition requires that protons, for example, should have kinetic energies larger than 2 MeV. For this reason, accurate experimental confirmations of the stopping-power theories had to await the advent of high-energy accelerators.

Strictly speaking, any change in velocity of a penetrating particle is associated with a change in direction and vice versa. For all practical purposes, however, one may treat these two as separate phenomena. The energy loss occurs predominantly via inelastic collisions with electrons, whereas large changes in direction are caused by infrequent Coulomb encounters with nuclei, with little energy transfer. To this approximation, one may consider each particle as moving along a well-defined and smooth track. Electrons of moderate energy are subject to multiple scattering by atomic electrons and hence move along piecewise smooth tracks, at best. Even here, the smooth track approximation is used as a simple and presumably also rather accurate expedient for stopping-power calculations.[13,14]

The result of the outlined calculation is the well-known BETHE formula for stopping powers:

$$-\frac{dE}{dx}=\frac{4\pi e^2(ez)^2}{mv^2}\times NZ\left[\log\frac{2mv^2}{I}+f_\mu(\beta^2)-\delta_{\text{pol}}\right] \qquad (2)$$

where (ez) and v are the charge and the velocity of the penetrating particle, m the electron rest mass, N the number of atoms per unit volume in the target, Z their atomic number, and $\beta = v/c$, where c is the velocity of light. The second term in parentheses of equation (2) accounts for the retardation at relativistic

velocities, and has different values for heavy particles, positrons and electrons. The third term takes into account energy-dependent (transversal) polarization effects on stopping. It is a strongly increasing function in v for $v \to c$; but $\delta_{\text{pol}} = 0$ for $v \lesssim c/\epsilon^{1/2}(0)$, where $\epsilon(0)$ denotes the static dielectric constant of the stopping medium. Transversal polarization effects will not be discussed in this survey.

The stopping power depends only on the velocity and charge of the moving particles but not on their mass. In regard to the stopping material, the mean excitation potential emerges as the key constant. It is defined as the product of all excitation levels in the substance, $\hbar\omega_{ik}$, each weighted exponentially by its respective oscillator strength, f_{ik}.

$$I = \hbar \prod \omega_{ik}^{f_{ik}} \tag{3}$$

where $\sum_{ik} f_{ik} = 1$.

A discussion of this important quantity will be given below.

4. ENERGY DEPENDENCE OF STOPPING POWERS AND RANGES

It may be informative at this point to exhibit typical stopping-power curves. Fig. 1 shows the energy dependence of the stopping power of paraffin for electrons and protons. At high energies, the stopping power, i.e. the energy transferred per unit tracklength, is small. It rises sharply over several orders of magnitude as the particles are slowed down. Eventually, the curves go through a maximum and tend towards zero. If these curves were drawn as a function of particle velocity they would nearly coincide. In terms of energy, however, the stopping power is lower for electrons than for heavy particles at low energies; the situation is reversed at high energies.

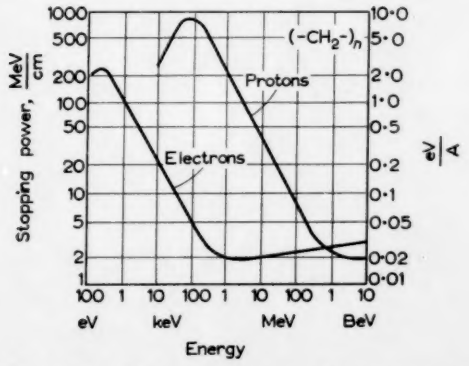


FIG. 1. Stopping power of polyethylene for electrons and protons versus particle energy.[36]

It may be noted parenthetically that this strong energy dependence has a particular bearing on chemical and biological radiation effects. Suppose an energy W has to be transferred on the average to the stopping medium in order to produce a certain effect. The density of such effects, n, will then be a linear function of the stopping power:

$$n = \frac{1}{W}\left(-\frac{dE}{dx}\right). \tag{4}$$

Since in general W is not known, one can study only the dependence of n on stopping power, under the tacit assumption that W is independent of the particle energy. Each event will affect a certain volume element of the material. As the stopping power exceeds a critical value, the affected volume elements start to overlap, and the effect may become stopping power dependent. The evidence for such stopping power dependence of chemical reactions has been discussed recently.[15,16] The stopping power dependence is also apparent in biological systems, where the biological effectiveness often changes as the energy transfer exceeds 2 eV/A,[17,18] i.e. the equivalent of one ionization potential per intermolecular distance.

Of considerable practical importance in dosimetry and many other fields are relative stopping powers and range energy relations. At equal densities and for moderate energies, the relative stopping power is given by

$$s = \frac{Z/A}{Z_r/A_r} \frac{\log 2mv^2 - \log I}{\log 2mv^2 - \log I_r} \tag{5}$$

where A is the atomic weight. The index r refers to the reference substance. It is apparent that with increasing v the function s increases or decreases monotonically, depending on whether I is larger or smaller than I_r or, equivalently, whether or not the reference substance is made of the lighter atoms. Equation (5) is well documented experimentally and illustrated in Fig. 2.

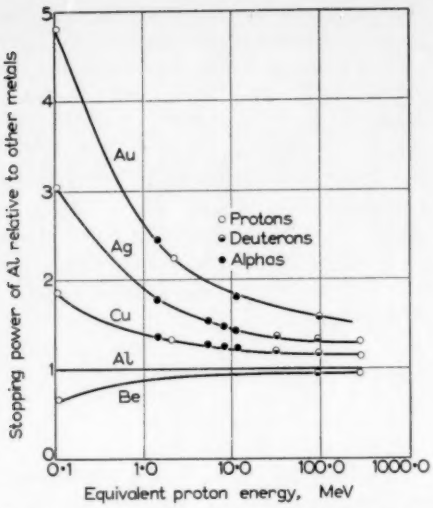


FIG. 2. Stopping power of aluminum relative to other metals (from data compiled by BETHE and ASHKIN[10]).

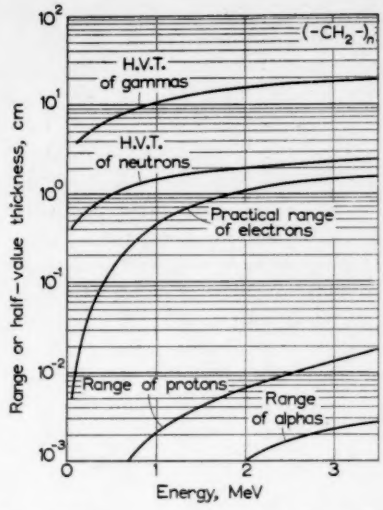


FIG. 3. Range and half-value thickness of ionizing radiation in polyethylene (after V. H. SUN, *Mod. Plastics* **82,** 141 (1954), based on G. R. WHITE, *Nat. Bur. Stand. Rep.* 1003 (1952); D. J. HUGHES *et al.*, AECU 2040 (1952); L. KATZ and A. PENFOLD, *Rev. Mod. Phys.* **24,** 28 (1952); W. BRANDT.[36]

The range of particles in a substance equals the energy integral over the inverse stopping power. Fig. 3 compares the range of heavy charged particles with the (practical) range of electrons and the half-value thicknesses of neutrons and X-rays in polyethylene. Fission fragments of approximately 100 MeV have ranges comparable to 3.5 MeV alpha particles. Clearly, radiation effects produced by corpuscular radiation occur with high density in thin layers; neutrons and X-rays affect large volumes in low densities.

5. MEAN EXCITATION POTENTIALS

The mean excitation potential I was stressed to be the only important material constant in the theory of stopping at particle velocities such that δ_{pol} remains small. Its definition, equation (3), offers a method for calculation in principle, which in practice can only be applied to the lightest elements. The mean excitation potential must therefore be used as an experimentally adjustable constant. Values for I are not only of much fundamental interest, they are of great practical importance in predicting stopping for one set of experimental conditions from another set, via equation (2).

BLOCH[4] has shown for the statistical model of the atom that in the Thomas–Fermi approximation:

$$I = KZ \qquad (6)$$

where K is an adjustable constant. Taking the

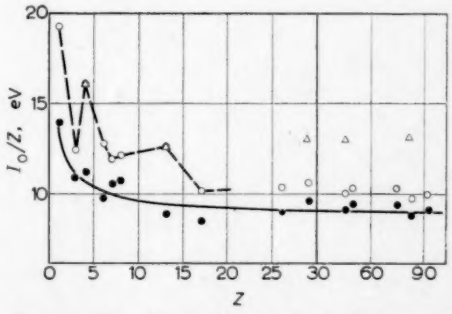


FIG. 4. Experimental mean excitation potentials and the statistical atomic model. ○: I/Z of bound atoms measured with 340 MeV protons and evaluated relative to $I_{Al} = 165$ eV;[20,31] ●: I_0/Z of isolated atoms.[28] △: I/Z measured with 10–20 MeV protons and corrected for inner shells.[24,26] The corrections apparently underestimate the stopping-power deficiencies of inner shells at low energies.

electron-spin interactions into account, the more accurate Thomas–Fermi–Dirac model predicts a Z-dependence for K also. JENSEN[19] finds approximately:

$$I_0 = K_0(1 + k_0 Z^{-2/3})Z. \qquad (7)$$

The index 0 refers to isolated atoms. The constants K_0, k_0 have not been calculated with any certainty except that k_0 should be <0.8; they must be determined experimentally.

In Fig. 4, I/Z values measured with 340 MeV protons[20,21] are plotted vs. Z (open circles). They confirm the expected gradual decrease of K with increasing Z, save for the superimposed fluctuations. As discussed below, these fluctuations largely are due to valence shell effects.

6. ENERGY DEPENDENCE OF MEAN EXCITATION POTENTIALS

For decreasing particle energies, BETHE's expression (2) becomes increasingly inaccurate, first for the transitions of electrons from K-shells, then from L-shells, and so on. Less energy can then be transferred to such electrons, and the stopping power decreases. If one determines I from low-energy data by equation (2), it appears velocity dependent. To preserve the inherent constancy of I, one makes corrections for the stopping-power deficiency of inner shells. Inner-shell corrections have been calculated for K- and L-shells by WALSKE,[22] using hydrogen-like wave functions. Judged by the importance of external shielding in different shells, they should be rather good approximations for K-shells, but less so for L-shells. At still lower energies, where the validity range of WALSKE's corrections is exceeded, one may follow ARON[23] and correct by omitting the stopping contributions of such shells altogether.

Values for I measured with low-energy protons agree with high-energy values for $Z \leq 13$, where only WALSKE's corrections need be employed (Fig. 4, triangles). They do *not* show a Z-dependent K for larger Z, where either WALSKE's corrections had to be extrapolated beyond their intended validity range[24,25] or ARON's corrections were used.[26] In other words, mean excitation potentials obtained at different energies show a slight but significant velocity dependence above and beyond the difference one accounts for by inner-shell corrections.

The Thomas–Fermi–Dirac model constitutes a fundamental improvement over the Thomas–Fermi model akin, say, to the Hartree–Fock approximation over the Hartree approximation in quantum mechanics. Therefore, sufficiently accurate experiments must find *some* decrease of K with increasing Z, if the statistical model has any validity for the theory of stopping. As shown in Fig. 4, only the available high-energy data show this Z-dependence. One is led to the conclusion that the inner-shell corrections presently applied *underestimate* the stopping-power deficiency of inner shells at low energies. Certainly, this would appear to be a plausible systematic error, but the problem needs careful theoretical attention. One estimates on the basis of the existing discrepancies that inner-shell corrections for 10–20 MeV protons overestimate I-values of heavy elements by ca. 30 per cent. This exceeds their stated uncertainty of ± 10 per cent.

Further evidence in support of equation (7) is provided by a Lindhard–Scharff diagram. The logarithmic term in equation (2) is called the electronic stopping number L. A plot of experimental L-values versus log $(v^2/v_0{}^2Z)$ (v_0 = normalization factor), which is the Thomas–Fermi approximation, breaks up in separate curves for different Z-ranges.[27] Only a diagram of L-values extrapolated to the case of isolated atoms, and with the abscissa in accord with the Thomas–Fermi–Dirac model, i.e. equal to log $(v^2/v_0{}^2Z(1 + k_0Z^{-2/3})$ [cf. equation (7)], yields the expected single smooth curve (Fig. 5).[28]

7. ADDITIVITY RULE

Electrons in inner shells are virtually unaffected by the chemical and physical states of atoms. Therefore it is to be expected that whenever the fraction of electrons in valence shells is small, as in heavy elements, the stopping power of compounds is given to a high approximation by the sum of the stopping powers of the component elements. This is called Bragg's rule. Slight deviations from Bragg's rule can occur whenever a significant fraction of the stopping electrons reside in valence shells, as

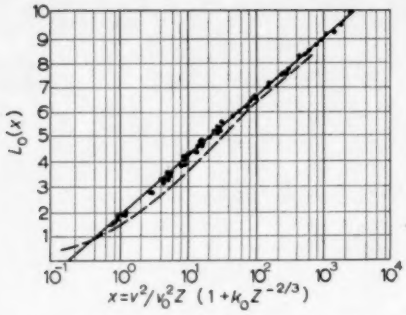


FIG. 5. Lindhard–Scharff diagram plotted for the Thomas–Fermi–Dirac model with $k_0 = 0.6$ as trial value.[28]

e.g. in organic compounds. Such deviations are of practical interest in cases where one wishes to measure the dose absorbed in tissues from the ionization in gases mixed so as to simulate the chemical tissue composition.

Bragg's rule must be exact for a dilute monoatomic gas composed of different atomic species in known stoichiometric proportions. The stopping power of such a "Bragg-gas" may be taken as the reference value.

Three causes for deviations from Bragg's rule may be distinguished: chemical binding, intermolecular interaction, and polarization. Consider an atom of the reference gas in its ground state (Fig. 6). After a chemical reaction has been induced, the atom will be in an electronic state which is *lower* than the free ground state owing to the quantum mechanical exchange forces which make chemical bonds stable. The resulting tighter binding of the valence electrons increases the mean excitation potential and hence reduces the amount of energy transferred during collision. The difference in I between different valence states should parallel in sign, but may exceed in magnitude the difference in bond energies per bonding electron.

The second effect on the stopping power of atoms stems from the intermolecular interactions in dense systems. Effects similar to the Stark effect and pressure broadening tend to change the energy levels of valence electrons. No direct information as to the sign and magnitude of these effects on stopping powers is available at present. Indirect information from refractive index studies and line shifts shows that in nonmetallic condensed phases a "red shift" may be anticipated because under ordinary circumstances the attractive forces between molecules in a lattice predominate and tend to loosen the electronic binding, whereas the repulsive exchange forces (which would give a "blue shift") are weak.

Finally, the field of a penetrating particle in condensed matter is shielded from interacting atoms by the polarization of the interjacent medium. The amount of shielding depends on the density of the substance ("density effect") and its dielectric properties. Since less energy can be transferred to shielded atoms, their effective stopping power is reduced in the condensed phase. Of course, each atom partly offsets this polarization reduction by the local field effect; the amount depends on the Lorentz factor of its environment.

The "zero-energy" (longitudinal) polarization effect has been taken into account only recently in the evaluation of mean excitation potentials[29] and their valence-shell dependence.[28] The situation arises frequently that the effect of shifting an atom from a valence state 1 to a valence state 2, with $I_1 < I_2$ (which *decreases* its stopping power), largely is balanced by a simultaneous overall decrease of the polarizability of the medium in which the atom is imbedded (which *increases* its effective stopping power), so that the additivity rule is obeyed more closely than is to be expected from valence

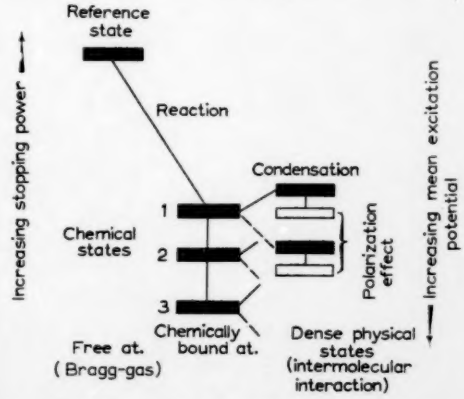


FIG. 6. Valence shell effects on stopping power (schematic).

Table 1. *Calculated mean excitation potentials of light elements**

Element	Valence state	I_{val} (eV)	Element	Valence state	I_{val} (eV)
H	Atomic	14	O	Atomic	77
	Molecular	18		Molecular	78
	Aliphatic	16		—O—	98
	Aromatic	19		O=	77
Li	Atomic	33	F	Atomic	86
	Metallic	38		Bound	87
Be	Atomic	42	Al	Atomic	120
	Metallic	60		Metallic	164
C	Atomic	60	P	Atomic	137
	Molecular (graphite)	71		Bound	147
	Aliphatic	80	S	Atomic	144
	Aromatic	71		Bound	171
N	Atomic	68	Cl	Atomic	154
	Molecular	78		Bound	183
	Amines, nitrates	73			
	In ring	90			

* Calculated according to ref. 28, based on $I_{Al} = 165$ eV as reference value and on equation (7) with $K_0 = 8.2$ eV and $k_0 = 0.70$.

state effects alone. Carbon in aromatic and aliphatic hydrocarbons is an example. In exceptional cases, the polarization effects may enhance the actual valence state effect. Hydrogen in the same compounds seems to present a case in point.

On the whole, deviations from Bragg's rule have long been estimated not to exceed 1–2 per cent in stopping.[9,10,30] More recently, a semitheoretical method was proposed to account for the various effects in some detail.[28] Suppose the I_0 value of an isolated atom is known. In accord with equation (3), I_{val} for a given valence state may be obtained through multiplying I_0 by the ratio of the mean excitation potentials of the valence electrons in the bound and the isolated states, assuming inner shells to remain unaffected. In practice, one can accomplish this to a good approximation through multiplying I_0 by the ratio of the respective mean optical absorption frequencies raised to a power equal to the fraction of electrons in valence shells. By inverting this method, I_0 values can be extracted from stopping-power experiments in condensed systems. One finds that such values fall on a smooth I_0/Z vs. Z curve (Fig. 4, solid circles) in agreement with equation (7) (Fig. 4, solid line). The best curve can be taken as the reference curve for I_0-values. Of course, the "best" numerical values for K_0 and k_0 may change somewhat as more experimental data become available. Also, the statistical model applies best to atoms with large Z, and its validity becomes increasingly uncertain as Z decreases. Fortunately, all empirical evidence attests that the statistical model has a higher *a posteriori* validity for light elements than could be expected from its very nature.

As presently best values are recommended $I(Al) = 165 \pm 3$ eV[24,26] for reference, $K_0 = 8.2$ eV and $k_0 = 0.70$ for calculating I_0 values by equation (7).[28] Table 1 lists some calculated I_{val} of light elements in various valence states, based on these values.

Table 2. Comparison of experimental and calculated stopping powers

Substance	Energy (MeV)	Stopping power (MeV/(g/cm²)) Experimental	Calculated‖	Deviation (%)
Polyethylene	270 (p)	4.01*	3.98	1
	5.9(α)	919†	913	1
Polyethylene teraphthalate	5.9(α)	736†	733	0
Polyvinyl alcohol	5.9(α)	848†	826	3
Aluminum	270 (p)	2.96‡	2.95	0
	18 (p)	21.6§	21.6	0
	5.9(α)	563†	558	1
Air	5.9(α)		720	

* THOMPSON.[31] ‡ BAKKER and SERGÈ[20] ‖ BRANDT.[36]
† PHELPS *et al.*[43] § SACHS and RICHARDSON.[44]

The magnitudes of valence shell effects are of interest. Upon bonding, I_0 increases in most cases by ca. 20 per cent, thereby reducing the stopping power by ca. 2 per cent. The differences between different bound states of light atoms are somewhat smaller. The polarization effect decreases the stopping power by varying amounts, ranging from 0.1 to 1 per cent. This also would be the maximum stopping power decrease to be observed on condensing a gas to a liquid or solid.

Many of the predicted details exceed the accuracy of the presently available experimental data, but the trends are in complete agreement with experiment, particularly with THOMPSON's very accurate data[31] and with the recent experiments on the effect of condensation by DE CARVALHO and YAGODA[32] and by ELLIS, ROSSI, and FAILLA.[33] The latter two investigations show that the stopping powers of water vapor and liquid are the same, at least within 2–5 per cent. They can not confirm APPLEYARD's result[34] that the liquid should have a (by present standards uninterpretable) 15 per cent higher stopping power than the vapor. The system $(CH)_{gas}$ vs. $(CH)_{solid}$ (acetylene vs. polystyrene)[35] also agrees with the theoretical expectations.

Table 2 lists stopping powers of some organic substances and of aluminum, calculated with the I_{val} values of Table 1 and the polarization correction discussed above.[36] Inner-shell corrections were accomplished by a cut-off procedure in the framework of the statistical model. The agreement with available experimental data evidently is good over a wide range of particle velocities.

BOHR[37] predicted large valence-state effects for light metals, which have since been confirmed quantitatively by experiment. The valence electrons in the metallic state interact weakly with the ion lattice, and their higher excitation levels bear only a faint resemblance to their energy levels in isolated atoms: they are essentially "free". Their stopping power results from the constraint imposed by their mutual Coulomb repulsion via collective motions and is hence a function of the electron density in the lattice. The situation is different for transition and noble metals where the conduction electrons interact more strongly with the ion cores, and, indeed, anomalies have been found for their stopping powers;[25,27,38,39] however, they can be reconciled with other known anomalies of these lattices.

PLATZMAN[9] has pointed out that alkali

halides should be another promising class of materials for studying valence state effects. The cognate LiH-crystal offers a particularly interesting example, because in the quantum theory of crystal bonding it holds an intermediate position between a covalent and an ionic lattice.[40,41] The resulting electron density distribution is as yet without experimental confirmation. One can estimate[42] that the mean excitation potential in a purely ionic LiH-crystal should be approximately twice as large as in a covalent crystal (~54 eV vs. ~28 eV); for the mixed but predominantly covalent bonding predicted by theory one finds an intermediate value (~39 eV). The respective calculated stopping powers differ so significantly that an appropriate stopping experiment could decide on the predominant bonding character of the LiH-crystal.

8. SUMMARY

This survey reviews the role played by various atomic factors in the stopping of charged particles in matter and discusses in some detail the velocity dependence of mean excitation potentials and the additivity rule for stopping powers. Many other subjects of current interest had to be omitted for brevity, such as the advances made during the last years in the stopping of low-energy heavy particles and of high-energy electrons, or the disparities in the current interpretations of characteristic energy losses of singly scattered electrons.

It is concluded that stopping powers can be calculated with high degree of accuracy over wide energy ranges. There still remains a discrepancy of ca. 2 per cent between the stopping powers calculated for heavy elements from high-energy experiments and those calculated from low-energy experiments. The discrepancy presumably arises from an underestimate of the stopping-power deficiency of inner shells at low energies as introduced by the presently applied inner-shell corrections. This conjecture is corroborated by the observation that only uncorrected mean excitation potentials measured at high energies concur with the A-dependence expected from the statistical Thomas–Fermi–Dirac model of the atom. Additional experimental and theoretical work is needed to resolve this point quantitatively. Calculated valence-state effects on stopping powers are confirmed by all recent experimental observations and can be included readily in stopping power calculations.

REFERENCES

1. E. Leithäuser, *Ann. Phys., Lpz.* **15,** 283–306 (1904).
2. N. Bohr, *Phil. Mag.* **25,** 10–31 (1913); *Ibid.* **30,** 581–612 (1915).
3. H. Bethe, *Ann. Phys., Lpz.* **5,** 325–400 (1930).
4. F. Bloch, *Z. Phys.* **81,** 363–376 (1933).
5. E. Fermi, *Phys. Rev.* **56,** 1242 (1939); *Ibid.* **57,** 485–493 (1940).
6. O. Halpern and H. Hall, *Phys. Rev.* **57,** 459–460 (1940); *Ibid.* **73,** 477–486 (1948).
7. G. C. Wick, *Nuovo. Cim.* **1,** 302–313 (1943).
8. R. M. Sternheimer, *Phys. Rev.* **88,** 851–859 (1952); *Ibid.* **103,** 511–515 (1956).
9. R. Platzman, *Symposium on Radiobiology* (Edited by J. J. Nickson) pp. 139–160. Wiley, New York (1952).
10. H. Bethe and J. Ashkin, *Passage of Radiations through Matter. Experimental Nuclear Physics* (Edited by E. Segrè) Vol. I, pp. 166–357. Wiley, New York (1953).
11. S. K. Allison and S. D. Warshaw, *Rev. Mod. Phys.* **25,** 779–817 (1953).
12. E. A. Uehling, *Ann. Rev. Nucl. Sci.* **4,** 315–350 (1954).
13. L. V. Spencer, *Phys. Rev.* **98,** 1597–1615 (1955).
14. A. T. Nelms, *Nat. Bur. Stand. Circ.* **577,** 1–30 (1956).
15. L. H. Gray, *J. Chim. Phys.* **52,** 519–527 (1955).
16. A. Chapiro, *Rad. Res.* **6,** 11–26 (1957).
17. R. E. Zirkle, *Radiation Biology* (Edited by A. Hollaender) Vol. I, pp. 315–350. McGraw-Hill, New York (1954).
18. J. B. Storer, P. S. Harris, J. E. Furchner and W. H. Langham, *Rad. Res.* **6,** 188–288 (1957).
19. H. Jensen, *Z. Phys.* **106,** 620–632 (1937).
20. C. E. Bakker and E. Segrè, *Phys. Rev.* **81,** 489–492 (1951).
21. R. Mather and E. Segrè, *Ibid.* **84,** 191–193 (1951).
22. C. M. Walske, *Phys. Rev.* **88,** 1283–1289 (1952); *Ibid.* **101,** 940–944 (1956).
23. W. A. Aron, *Univ. Calif. Rad. Lab. Rep UCRL* 1325, 1–46. Unpublished (1951).
24. D. O. Caldwell, *Phys. Rev.* **100,** 291–294 (1955).
25. V. C. Burkig and K. R. MacKenzie, *Phys. Rev.* **106,** 848–851 (1957).

26. H. Bichsel, R. F. Mozley and W. A. Aron, *Phys. Rev.* **105,** 1788–1795 (1957).
27. C. P. Sonett and K. R. MacKenzie, *Phys. Rev.* **100,** 734–737 (1955).
28. W. Brandt, *Phys. Rev.* **104,** 691–693 (1956); *Ibid.* **105,** 1933 (1957)(E).
29. R. M. Sternheimer, *Phys. Rev.* **93,** 351 (1954).
30. L. H. Gray, *Proc. Camb. Phil. Soc.* **40,** 72–102 (1944).
31. T. Thompson, Effect of chemical structure on stopping powers for high-energy protons. Thesis, Univ. Calif. Unpublished (1952).
32. H. G. de Carvalho and H. Yagoda, *Phys. Rev.* **88,** 273–278 (1952).
33. R. H. Ellis, H. H. Rossi and G. Failla, *Phys. Rev.* **97,** 1043–1047 (1955).
34. R. K. Appleyard, *Nature, Lond.* **163,** 526 (1949); *Proc. Camb. Phil. Soc.* **47,** 443–445 (1951).
35. R. H. Ellis, H. H. Rossi and G. Failla, *Phys. Rev.* **86,** 562–563 (1952).
36. W. Brandt, *Du Pont Rad. Phys. Lab. Res. Rep.* Unpublished (1957)
37. A. Bohr, *Kgl. Danske Vidensk. Selsk. Mat.-fys. Medd.* **25,** 19, 1–52 (1948).
38. D. W. Green, J. N. Cooper and J. C. Harris, *Phys. Rev.* **98,** 466–470 (1955).
39. M. Bader, R. E. Pixley, F. S. Moser and W. Whaling, *Phys. Rev.* **103,** 32–38 (1956).
40. D. H. Ewing and F. Seitz, *Phys. Rev.* **50,** 760–777 (1936).
41. S. O. Lundquist, *Ark. Fys.* **8,** 177–196 (1954).
42. W. Brandt, *Physik. Verh.* **8,** 188–189 (1957).
43. J. Phelps, W. F. Huebner and F. Hutchinson, *Phys. Rev.* **95,** 441–444 (1954).
44. S. C. Sachs and J. R. Richardson, *Phys. Rev.* **89,** 1163–1164 (1953).

Health Physics Pergamon Press 1958. Vol. 1. pp. 21–26

A SINGLE ION DETECTOR FOR MEASUREMENT OF γ-RAY IONIZATION IN CAVITIES

J. A. AUXIER, G. S. HURST and R. E. ZEDLER
Health Physics Division, Oak Ridge National Laboratory, Operated by Union Carbide Nuclear Company for the Atomic Energy Commission, Oak Ridge, Tennessee

(*Received* 11 *December* 1957)

Abstract—A new method of measuring the energy absorbed in a medium due to γ-ray irradiation is described. The method requires that a counting device have linear dimensions which are small compared to the mean free path for ionization of the counting gas by the secondary electrons. Under this condition the number of ionizing events in the counter is proportional to the energy absorbed in the gas; thus the energy absorbed in the surrounding wall may be inferred from the count rate by means of the Bragg–Gray principle. Application of the method is made to a tissue dosimeter which is capable of measuring the dose due to γ-rays even when mixed with fast neutrons.

1. INTRODUCTION

THE use of cavities to measure the energy absorbed in tissue from γ-radiation prescribes that the cavity be surrounded with materials having a low Z.[1] Consequently fast neutrons may transfer appreciable energy to these materials by elastic collisions. It has been shown[2] that one tissue rad of fast neutrons will transfer from 0.1 to 0.5 rad to graphite depending on the neutron energy (in the range of 0.1–20 MeV). If the amount of ionization produced in a small cavity inside a graphite block were measured with the usual ionization current method, the ionization f_i due to one tissue rad of neutrons relative to the ionization due to one tissue rad of γ-rays would be given by

$$f_i = F \cdot W_e/W_a^* \qquad (1)$$

where F is the ratio of the first collision dose in graphite to the first collision dose in tissue for the same neutron field, W_e is the average energy dissipated per ion pair formed in the gas by electrons, and W_a is the average W value for the recoil atoms. Thus, if $W_e = W_a$, the neutron response of the ionization chamber would range between 10 and 50 per cent. Actually, W_a is very poorly known, hence f_i cannot be calculated with certainty.

* Assuming the cavity gas has the same atomic composition as the wall.

2. THEORY OF THE SINGLE ION DETECTOR

The search for a method of measuring ionization which could distinguish between that produced by the secondary electrons, generated by γ-rays, from that produced by recoil atoms led to the following considerations. Suppose that the linear dimension, t, of a cavity is very small and the gas pressure in the cavity is low so that the mean free path for ionization by electrons, λ_e, is much greater than t. This is equivalent to the statement that the probability of the electron making an ionizing collision in the gas is low. The probability of creating n ion pairs per traversal will be given by the Poisson distribution,

$$P_n = \frac{1}{n!}\left(\frac{t}{\lambda_e}\right)^n e^{-t/\lambda_e}. \qquad (2)$$

In the case cited above $t/\lambda_e \ll 1$, and hence $P_1 = t/\lambda_e$ and

$$P_n = \frac{1}{n!}(P_1)^n. \qquad (3)$$

As an example, suppose $P_1 = 0.1$, then the probability of two ion pairs, $P_2 = 0.005$.

Similarly the probability $P_n \ll P_1 \quad (n > 2)$, hence to a good approximation each ionizing event leaves behind one ion pair. If each ionization can be detected then the number of counts equals the number of ion pairs and the energy absorbed in the cavity is simply W_e times the number of counts.

On the other hand, for fast neutron recoils the mean free path for ionization, λ_a, may be much smaller than t under the same condition where $\lambda_e \gg t$, corresponding to the creating of a large average number of electrons $\bar{N} = t/\lambda_a$ per traversal. The neutron response of the single ion detector as compared to γ-radiation is now given by

$$f_{\text{sld}} = f_i/\bar{N} = \frac{W_e}{W_a} \times \frac{F}{\bar{N}}. \tag{4}$$

But, $\bar{N} = t/\lambda_a = \dfrac{t(dE/dx)_a}{W_a}$; hence,

$$f_{\text{sld}} = \frac{W_e}{W_a} \times \frac{W_a}{t(dE/dx)_a} \times F = \frac{W_e F}{t(dE/dx)_a} \tag{5}$$

where $(dE/dx)_a$ is the stopping power for the recoil atoms. It is interesting to note that the poorly known quantity W_a is eliminated.

3. CONSIDERATION OF γ-RAY ENERGY RESPONSE

Consider the energy response of the counter. Let E_γ = energy of the incident γ-rays; $N(E)$ = secondary electron spectrum produced in the medium by the γ-rays of energy E_γ; and ϵ = energy absorbed in the cavity. Then,

$$\epsilon = W_e \int_0^\infty P_1(E)\, N(E)\, dE + 2W_e \times \int_0^\infty P_2(E)\, N(E)\, dE + \ldots nW_e \times \int_0^\infty P_n(E)\, N(E)\, dE \tag{6}$$

while the actual number of counts is given by

$$C = \int_0^\infty P_1(E)\, N(E)\, dE + \int_0^\infty P_2(E)\, N(E)\, dE + \ldots \int_0^\infty P_n(E)\, N(E)\, dE. \tag{7}$$

The ratio C/ϵ gives the "response" of the counter, and hence we are interested in the dependence of C/ϵ on E_γ. In the approximation above (i.e. $P_2 \ll P_1$) C/ϵ is just $1/W$. But this may not hold even when E_γ is in the MeV region, since the total number of electrons in the part of the energy spectrum ($N(E)$) for which P_2 is not small compared to P_1 may not be negligible. This effect is, in fact, suggested by the cloud chamber work of WILSON,[3] some interesting parts of which are quoted by LEA[4] where he mentions that clustering (two or more electron charges on a single droplet) may account for 57 per cent of the ionization in some cases.*

Unfortunately, in carrying out the computation of C/ϵ, $N(E)$ is unavailable[5] when $E_\gamma < 50$ keV. Additional difficulties arise in the evaluation of the P's. The problem in this connection is finding λ_e, the mean free path for electron ionization by collision.

For a one-medium problem, we may write

$$\lambda_e = W_e/(dE/dx)_e \tag{8}$$

where $(dE/dx)_e$ is the stopping power. However, as pointed out by SPENCER and ATTIX,[6] for two media, $(dE/dx)_e$ is not easily evaluated. They have treated a schematic problem involving a parameter, Δ, which is the effective cavity dimensions in electron energy units. Unfortunately, for a cavity of the size being considered here, the value of Δ is unknown.

When the gas in the cavity is at a very low pressure, still another complication must be considered. TAYLOR[7] quotes work which showed that the saturation curve for a cylindrical vacuum ionization chamber was grossly unsymmetrical with respect to direction of electrical field. This has been interpreted by GREENING[8] as being due to the dependence of the collection efficiency of the low energy electrons liberated from the wall material on the electric field direction and surface areas of the electrodes. Therefore, the electrical field must play an important role in determining the amount of ionization collected from a "near vacuum" chamber. Even the value of Δ in the Spencer–Attix theory will depend on the field.

* This effect was kindly called to our attention in conversation with L. H. GRAY.

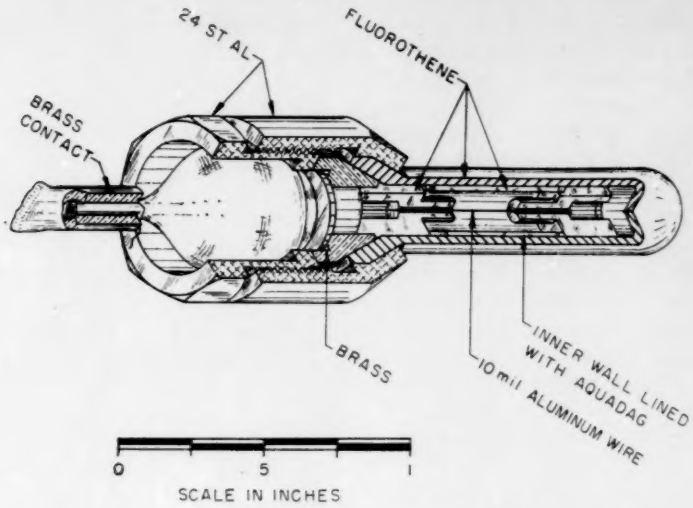


FIG. 1. Cutaway view of detector.

This is, in fact, a condition which should be imposed on the Spencer–Attix theory and is especially important for any geometry other than parallel plane electrodes.

Because of the lack of information, mainly in knowledge of $(dE/dx)_e$ at low energy, it is not possible to predict the performance of the proposed low pressure counter; consequently the conclusions which are drawn will be based on empirical results as reported in the next section. However, there is no reason to suspect that the remarks made about the low neutron sensitivity will not hold.

4. DESCRIPTION OF THE COUNTER

(a) *Design*

As pointed out in connection with equation (8), $(dE/dx)_e$ as determined from the BETHE formula[9] is not appropriate for determining the mean free path for ionization for cavities of the size considered here. Since $(dE/dx)_e$ is almost entirely unknown below 50 keV electron energy, it was decided to base the design on primary specific ionization data rather than determining λ_e from equation (8). It was assumed that an isotropic distribution of electrons existed in the wall materials and the average traversal of electrons to be approximately four-thirds times the radius.[10,11] If the primary specific ionization at 20 keV electron energy as quoted by WILKINSON[12] is used, then P_1 is about 0.1 for a cavity of 3 mm average traversal filled to a pressure of air or CO_2 at 2.25 mm of Hg. For comparison, if values of $(dE/dx)_e$ are put into equation (8), the corresponding pressure is 0.5 mm Hg. The design which will be discussed below represents a compromise and uses 1 mm Hg pressure of gas in a 0.5 cm diameter cavity.

The design of a working model of the detector is shown in Fig. 1. The sensitive volume is a right cylinder 0.5 × 0.5 cm. The cylinder walls are materials of intermediate Z, such as graphite, fluorethene, or aluminum, and the center electrode is wire 0.0254 cm, which passes through a fluorethene insulator and a kovar seal. The wire and a capillary copper tube of desired length are soldered to the kovar terminal such that the passage of gas into or out of the counter is not blocked. When the detector is filled with gas the length of copper tubing is sealed off at the distal end by crimping and soldering. For cases where the wall materials are permeable to gas, a thin gas-tight aluminum shell is used to cover the wall, and for a non-conducting wall a thin application of graphite is made to make it conducting.

Several gases have been used, including air, carbon disulfide, carbon dioxide, and *iso*butane. Although *iso*butane is a hydrogenous gas,

the total mass of hydrogen is so small that it contributes a negligible amount to the neutron response. All of these gases function to some degree, but CO_2 is more stable than air or CS_2, and *iso*butane is still more stable. A counter filled with *iso*butane will show negligible drift during weeks of operation. All of the gases operate well at 1 mm of Hg pressure, but *iso*-butane operates well at pressures as low as 0.75 mm of Hg. Except for stability the operating characteristics of the various gases are quite similar; only CO_2 and *iso*butane will be considered further in this paper.

Now that the design of the counter has been prescribed, its response to fast neutrons may be estimated, using equation (5). Since carbon is the element of lowest Z contained in the counter (except for the minute quantity of hydrogen in the *iso*butane gas) it will be assumed that the neutron response is due to carbon recoils. Table 1 shows the results of the calculation for various neutron energies, E_n. The results of SNYDER and NEUFELD[13] for carbon ions in tissue are used for the quantity $(dE/dx)_a$ and are listed for various values of the average energy $\bar{E}_c$ of the carbon recoils.

Table 1. Calculation of the neutron response of the single ionization detector

E_n	F (%)	$\bar{E}_c$ (MeV)	$(dE/dx)_a$ (keV/cm)	f_{sid} (%)
1.0	14.9	0.14	375	0.32
2.0	14.5	0.28	600	0.20
3.0	15.1	0.42	1080	0.15
4.0	24.7	0.56	1600	0.12
5.0	16.8	0.70	2000	0.07
10.0	34.1	1.4	3200	0.085
20.0	48.7	2.8	4800	0.081

(b) *Operation characteristics*

In operation the counter is connected to a preamplifier (gain of 20), amplifier (gain up to 10,000), and high voltage system. A well regulated low ripple anode voltage supply is used at approximately 500–600 V.

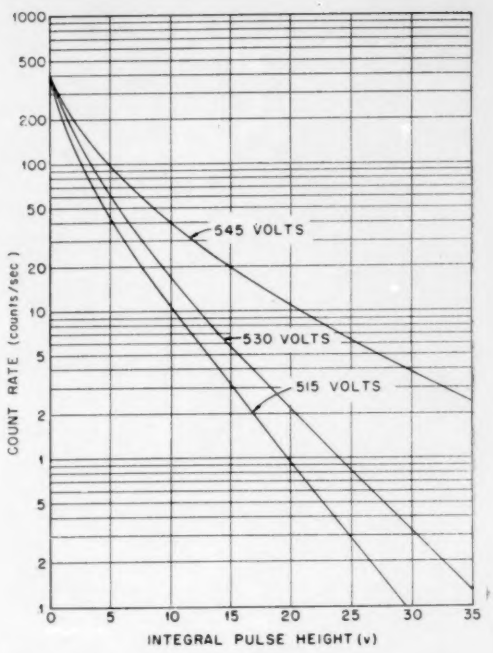


FIG. 2. Pulse height distributions for various values of anode voltage.

Fig. 2 depicts a family of pulse height distributions (i.e. count rate versus integral pulse height) as a function of anode voltage for the case of the detector in a Co^{60} γ-ray field of constant intensity.

Because of the widely separated probabilities for the formation of 1, 2, 3, etc. ion pairs, one would expect that the pulse height distribution would have sharp breaks at pulse heights corresponding to 1, 2, 3, etc. ion pairs. This structure is not observed experimentally, probably indicating that the counter has very poor resolution. This effect, in turn, is most likely due to the existence of gas amplification throughout the counting volume. Although there is no structure in the distribution for any gas or applied voltage, the curves approach a common value as the pulse height values decrease to the electronic noise level. The detector may be calibrated at any anode voltage and pulse height yielding acceptable sensitivity, but the energy response is more nearly correct when used at the lower pulse height values and at the

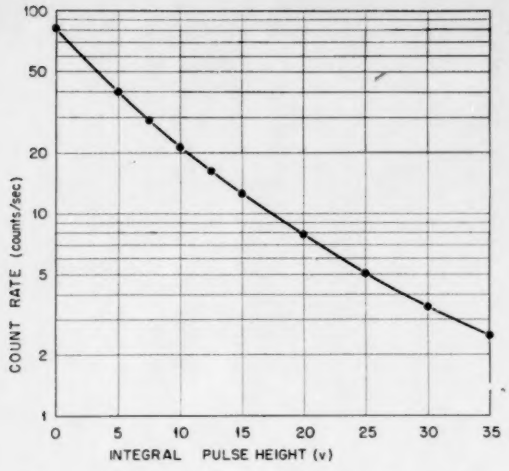


FIG. 3. Pulse height distribution for various sources of radiation.

highest anode voltage at which the counter will remain stable for extended periods of time.

Fig. 3 shows the pulse height distribution for γ-rays from Co^{60} (0.50 r/hr) and Cs^{137} (0.5 r/hr), and the associated γ-rays from a Po-Be fast neutron source (2.0 tissue rads/hr neutron dose-rate).

At low X- and γ-ray energies, the relative response to a fixed dose rate is greater than at higher energies, even with best operating conditions. The response of a typical detector is given in Fig. 4 for several thicknesses of tin surrounding the cavity. By putting tin foil around the counter, the response was made to equal that of the ionization chambers to much lower energies. Due to the very small mass of gas in the chamber (about 10^{-6} g) the ejection of photoelectrons from the anode was considered a source of increased response at low γ-ray energy and the anode subsequently was changed from stainless steel to aluminum, the dimensions remaining the same. This improved the low energy response, but not sufficiently to permit removal of all the tin, Further refinement of the detector, especially as concerns removal of other higher Z materials (i.e. the kovar seal) should also improve the response curve.

Gamma-ray measurements were also made with Lauritzen electroscopes for energies greater than 200 keV, and with condenser r-meters at energies less than 200 keV. The condenser r-meters had been compared to a standard air chamber,[14] for a range of values of effective X-ray energy extending below those shown in Fig. 4 (50 keV). The effective energy of a heterochromatic X-ray beam is the energy of a monochromatic beam which has the same absorption coefficient as the given beam in an incremental thickness of standard filter material. The International Committee on Radiological Units in 1937 made recommendations for standard filters.[15] The results were comparable to those of DAY,[16] who found a ratio of standard chamber reading to r-meter reading of about 1.01 at an applied potential of 75 kV (50/kV effective when using 6 mm Al filtration). With an applied potential of 50 kV, Day found the ratio to be only 1.03. Therefore, within the experimental limits of accuracy, Fig. 4 shows the response of the single ion detector relative to the standard air chamber.

The operating range of the instrument is largely prescribed by the time that can be used for counting for the low dose rates, and the speed of the electronics for the high dose rates. It is

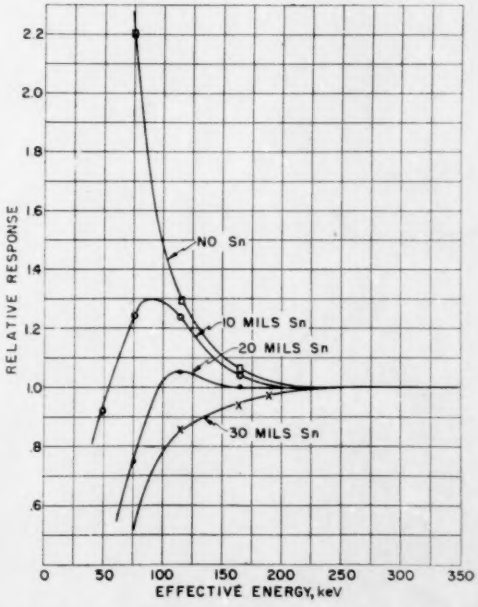


FIG. 4. Relative response as a function of energy.

approximately 20 mr/hr to 50 r/hr as the instrument has been used in its present form.

As already mentioned the detector was exposed to a Po–Be neutron source, and the response was as shown in Fig. 3. For the neutron source, the curve has the same slope as the gamma curves at all points below 35 V, and has been normalized to coincide with them. However, at pulse heights above 35 V, where the gamma pulses are sensibly cut off, there are, as can be seen on an oscilloscope, occasional large pulses that can be attributed to heavy ion recoils in the cavity. However, the count rate due to these large pulses is too small to distort appreciably the pulse height curve at pulse height values much below 35 V, except for the case of too low anode voltage which results in further lowering the γ-ray cut-off point (Fig. 5). For best operating conditions, assuming all Po–Be source response to be due to γ-rays, the ratio of γ-ray absorbed dose to the fast neutron absorbed dose is about 0.25. The calculated value of the same ratio of Po–Be sources based on the gamma-ray spectrum reported by BREEN, HERTZ and WRIGHT[17] and BREEN and HERTZ[18] is 0.23. This difference is less than experimental error, confirming the fact that the fast neutron response of the counter is negligible.

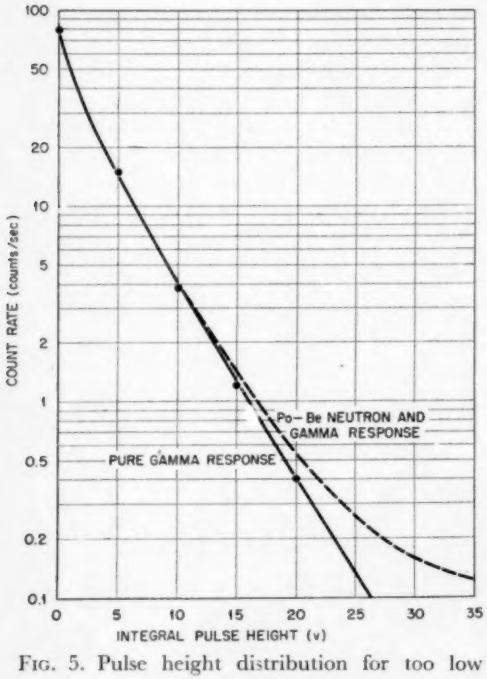


FIG. 5. Pulse height distribution for too low anode voltage.

REFERENCES

1. L. H. GRAY, *Proc. Roy. Soc.* A **156,** 578–596 (1936).
2. G. S. HURST, W. A. MILLS, F. P. CONTE and A. C. UPTON, *Rad. Res.* **4,** No. 1, 49 (1956).
3. C. T. R. WILSON, *Proc. Roy. Soc.* A **104,** 1–24 (1923); *Ibid.* A **104,** 192–212 (1923).
4. D. E. LEA, *Actions of Radiations on Living Cells.* Cambridge University Press (1947).
5. L. V. SPENCER and U. FANO, *Phys. Rev.* **93,** 1172 (1954).
6. L. V. SPENCER, and F. H. ATTIX, *Rad. Res.* **3,** 239 (1955).
7. L. S. TAYLOR, *Brit. J. Radiol.* **24,** No. 278, 67 (1951).
8. J. R. GREENING, *Brit. J. Radiol.* **27,** No. 315, 163–170 (1954).
9. H. A. BETHE, *Handbuch der physik* Vol. 24, Part I, p. 491. Springer, Berlin (1933).
10. F. H. MURRAY, CP–2922, p. 13 (1945).
11. S. I. TOMKEIEFF, *Nature, Lond.* **155,** 24 (1945).
12. D. H. WILKINSON, *Ionization Chambers and Counters.* Cambridge University Press (1950).
13. W. S. SNYDER and J. NEUFELD, *Rad. Res.* **6,** 67–78 (1957).
14. H. H. HUBBELL, Health Physics Division, Oak Ridge National Laboratory. Private communication.
15. GLASSER, QURMBY, TAYLOR and WEATHERWAX, *Physical Foundations of Radiology* (2nd Ed.) p. 244. Hoeber, New York (1952).
16. F. H. DAY, *NBS Research Paper* RP 1926, Vol. 41, p. 297, Fig. 1 (1948).
17. R. J. BREEN, M. R. HERTZ and D. U. WRIGHT, JR., MLM 1054, p. 24 (1954).
18. R. J. BREEN and M. R. HERTZ, *Phys. Rev.* **98,** No. 3, 599 (1955).

Health Physics Pergamon Press 1958. Vol. 1. pp. 27–33

SPECTRAL DISTRIBUTION OF ELECTRON FLUX IN A BETA-RADIOACTIVE MEDIUM

R. D. BIRKHOFF, H. H. HUBBELL, JR., J. S. CHEKA and R. H. RITCHIE

Health Physics Division, Oak Ridge National Laboratory
Operated by Union Carbide Nuclear Company for the Atomic Energy Commission
Oak Ridge, Tennessee

(*Received* 1 *November* 1957)

Abstract—Beta rays from a sample of P^{32} uniformly dispersed in a bakelite medium of size larger than the beta range produce secondary electrons as they slow down. The total flux including both primary and secondary electrons may be calculated from the Spencer–Fano–Attix theory and may be compared with the flux emerging from a channel into a vacuum cavity in the medium. Measurements made with a magnetic spectrometer show good agreement with the theory in the energy region above 0.35 MeV. Slight disagreement in the region 0.05 to 0.35 MeV is not considered serious because of approximations made to the exact theory and because of loss of spectrometer transmission at these lower energies.

INTRODUCTION

A complete understanding of the effects of radiation on biological material depends in part upon a detailed knowledge of the energy distribution of the electrons generated in the material by the action of ionizing radiation. Whatever the source of radiation (X-rays, β-rays, fast neutrons, etc.), one end product, from a physical standpoint, of the interaction of the radiation and the medium is the production of secondary electrons by ionizing collisions of the primaries (Compton electrons, β-rays, etc.), and such secondaries doubtless contribute significantly to the total radiation damage. For instance, GREENING[1] has estimated that about 97 per cent of the electrons liberated in a medium by the passage of energetic radiation have energies less than 100 eV, and electrons in this energy range are known to interact strongly with the atoms and molecules of a medium.

In the work described here the electron flux present in a mixture of P^{32} and bakelite plastic is measured over the energy interval from 1.7 MeV to 50 keV. A channel cut through the medium to a vacuum cavity lying within the medium allows a sample of the electron flux to escape into a beta spectrometer for energy analysis. This method of measurement is based on a remark by SPENCER and FANO[2] in a paper in which a method for calculating the flux is given. A simplified treatment of this theory has been given by SPENCER and ATTIX[3] and is reviewed here.

The inability of the beta spectrometer used here to transmit electrons of much less than 50 keV limited the investigation to energies above this value. The experimental problems involved in making measurements at lower energies are similar to those found in the investigation of the shapes of beta spectra at low energies, and these problems have yet to be solved in a completely satisfactory manner. There is also a limitation on the accuracy of the theory of the electron flux spectrum due to a lack of information on the stopping power and atomic ionization cross-section at low energies. Thus both a measurement of and a calculation of the most interesting portion of the electron flux spectrum, that which lies below several hundred electron volts, must await progress in both experimental and theoretical techniques.

THE FLUX OF PRIMARY ELECTRONS

An approximation to the Spencer–Fano theory of electron slowing down has been

obtained by SPENCER and ATTIX, and will be followed here. For simplicity the time factor is omitted in the following discussion. Thus the functions S and y may be assumed to represent the track density and flux integrated over some period of observation.

Let there be S electrons per cm^3 produced by the source in the medium having initial energies greater than a particular energy T. Then there will be S electron track segments per cm^3 of length Δx along which each electron drops in energy from T to $T - \Delta T$. There will also be S points per cm^3 in the medium corresponding to the electrons of energy T.

Consider a spherical probe in the medium having a cross-sectional area of 1 cm^2. It is desired to calculate the total number of electron tracks which will cut the sphere for which the associated electron energy will be between T and $T - \Delta T$. If one surrounds the unit sphere with a spherical shell of radius r, the number of points corresponding to electrons of energy T which lie within a distance between r and $r + dr$ of the unit sphere is just $S \times 4\pi r^2 dr$.

If the electron tracks emanate isotropically from each point in the spherical shell, and the probe subtends a fractional solid angle of $(4\pi r^2)^{-1}$ for each such point, then the total number of tracks which will originate in the shell and intercept the probe is

$$S \times 4\pi r^2 (4\pi r^2)^{-1}\, dr = S\, dr.$$

In order to calculate the total number of tracks corresponding to energies between T and $T - \Delta T$ which can intercept the shell from all distances r from the probe, it must be realized that the tracks must originate at distances no greater than Δx where

$$\Delta x = \frac{\Delta T}{(dT/dx)(T)}.$$

The total number of such tracks is then

$$\int_0^{\Delta x} S\, dr = S\, \Delta x \frac{\text{electrons}}{\text{cm}^2}.$$

The above expression may be equated to the flux crossing the probe with energy between T, $T - \Delta T$.

$$S\Delta x = y(T)\Delta T.$$

In the limit then and considering only the primary electrons

$$y(T) = \left[\frac{dT}{dx}(T)\right]^{-1} \int_T^{T_0} N(T')\, dT' \qquad (1)$$

where (NT') is the spatial density of the primary beta spectrum, and (dT/dx) (T) is the stopping power of the absorbing material for electrons of energy T.

Equation (3) may be obtained also from consideration of energy conservation. The absorption of energy per unit volume from electrons of energy between T and $T + dT$ is simply the product of the electron flux and the stopping power. That is

$$E_{ab} = y(T) \frac{dT}{dx}(T)\, dT. \qquad (2)$$

The density of electrons in the medium having energies in the interval T, $T + dT$ is the integral of the nuclear beta spectrum from T to the end point energy T_0. That is, all electrons of energy greater than T will at some point along their tracks pass through the energy interval between T and $T + dT$. The energy input into this interval is then the product of this electron density and the interval width

$$E_{in} = dT \int_T^{T_0} N(T')\, dT' \qquad (3)$$

The equating of E_{in} to E_{ab} then yields equation (1).

THE BUILD-UP OF SECONDARY ELECTRONS

Ionizing collisions of the primary electrons considered above will result in a large number of low energy secondary electrons in the medium. It is desired to calculate the number of such electrons having energy greater than T. This number may then be added to the integral of equation (1) and the total divided by the stopping power to obtain $y(T)$.

The probability per unit path for a transfer of energy τ from a primary electron of energy T' has been given by MÖLLER[4] as

$$k_M(T',\tau) = \frac{2\pi N_e r_0^2}{(\beta')^2}\left\{\frac{1}{\tau^2} + \frac{1}{(T'-\tau)^2} - [(2 + (T')^{-1})\,(T'+1)^{-2}] \times [\tau^{-1} + (T'-\tau)^{-1}] + (T'+1)^{-2}\right\} \qquad (4)$$

where N_e is the electron density, r_0 is the classical electron radius, $\beta' = v'/c$, and all energies are in mc^2 units. The problem is to calculate the number of secondaries of energy greater than T produced by a primary of energy T', i.e. the number of electrons which have an energy between $T'/2$ and T in accordance with the convention that the less energetic electron emerging from a collision is to be called the secondary. The relationships among the various energies may be visualized as in the energy level diagram of Fig. 1. Denoting the probability per unit path that an electron is created in the cross-hatched region from a primary at T' by $K(T', T)$, then

$$K(T', T) = \int_{T'/2}^{T'-T} k_M(T', \tau)\, d\tau. \tag{5}$$

Multiplying $K(T', T)$ by the flux at T' and integrating over possible values of T' from $2T$ to the highest primary energy T_0, the number of secondaries above T is found to be:

$$\int_{2T}^{T_0} K(T', T) y(T')\, dT'. \tag{6}$$

When the number of secondaries is added to the number of primaries above T, the final expression for the flux is

$$y(T) = \left[\frac{dT}{dx}(T)\right]^{-1} \left[\int_T^{T_0} N(T')\, dT' + \int_{2T}^{T_0} K(T', T) y(T')\, dT'\right]. \tag{7}$$

NUMERICAL EVALUATION OF THE FLUX

The calculation of the flux spectrum proceeds from the integral equation (7) by straight forward numerical methods. The integrated Möller formula with energies in mc^2 units may be approximated as

$$K(T', T) = \frac{2\pi N_e r_0^2}{(\beta')^2}\left[\frac{1}{T} + \frac{1}{T - T'}\right] \equiv H(T')\left[\frac{1}{T} + \frac{1}{T - T'}\right]. \tag{8}$$

The Bethe[5] stopping power for a polyatomic substance may be written

$$\frac{dT}{dx}(T) = \frac{2\pi N_A r_0^2}{\beta^2}\, \rho \sum_i \frac{f_i Z_i}{A_i}\left\{\ln \frac{\beta^2 T}{2(1-\beta^2) I_i^2} - (2\sqrt{(1-\beta^2)} - 1 + \beta^2)\ln 2 + (1-\beta^2) + \frac{1}{8}(1 - \sqrt{(1-\beta^2)})^2\right\} \tag{9}$$

where N_A is Avogadro's number, ρ is the density, and Z_i, f_i, I_i, and A_i are the atomic numbers, fractional mass abundances, average ionization potentials, and atomic weights, respectively, of the constituents of the absorbing material. Equation (9) may be written:

$$\frac{dT}{dx}(T) = \frac{2\pi N_e r_0^2}{\beta^2}\, \rho \frac{N_A}{N_e}\left[\sum \frac{f_i Z_i}{A_i} \times \left\{-(2\sqrt{(1-\beta^2)} - 1 + \beta^2)\ln 2 + (1-\beta^2) + \frac{1}{8}(1 - \sqrt{(1-\beta^2)})^2 + \ln \frac{\beta^2 \left[\frac{1}{\sqrt{(1-\beta^2)} - 1}\right]}{1 - \beta^3}\right\} + \sum \frac{f_i Z_i}{A_i} \ln \frac{1}{2 I_i^2}\right] = H(T)\left[f(\beta) + \frac{\sum \frac{f_i Z_i}{A_i} \ln \frac{1}{2 I_i^2}}{\sum \frac{f_i Z_i}{A_i}}\right] \equiv H(T) B(T) \tag{10}$$

where $f(\beta)$ is the quantity in braces $\{\}$.

Equation (7) may be simplified if a function $R(T)$ is defined such that:

$$R(T) = H(T)\, y(T) \tag{11}$$

Then equation (7) becomes:

$$R(T) = B^{-1}(T)\left\{P(T) + \int_{2T}^{T_0}\left[\frac{T'}{T} + \frac{T'}{T - T'}\right] R(T')\, \frac{dT'}{T'}\right\}. \tag{12}$$

Here $P(T)$ represents the integral primary spectrum. The solution of the equation is obtained for energies equally spaced on a logarithmic scale. For this particular problem the interval was chosen such that

$$T_n = T_0\, 2^{-n/5} \tag{13}$$

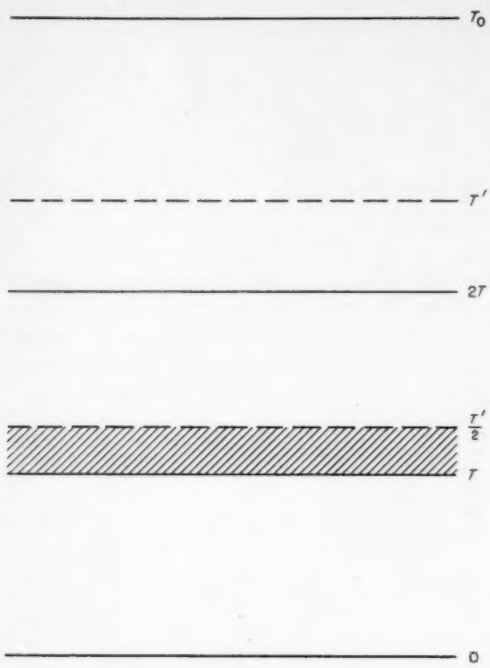


FIG. 1. Energy level diagram for production of secondary electrons.

Introducing logarithmic energy variables $\mu = \ln \frac{T_0}{T}$ and $\mu' = \ln \frac{T_0}{T'}$,

Equation (12) becomes

$$R(T_0, \mu) = B^{-1}(T_0, \mu) \left\{ P(T_0, \mu) + \int_0^{\mu - \ln 2} \left[\exp(\mu - \mu') + (\exp(\mu' - \mu) - 1)^{-1} \right] \times R(T_0, \mu')\, d\mu' \right\}. \quad (14)$$

The solution to the integral equation (14) may be obtained by approximating the integral by finite sums. For an integration interval $\Delta\mu = \frac{1}{5} \ln 2$ equation (14) then becomes:

$$R_n = B_n^{-1} \left\{ P_n + \sum_{m=0}^{m=n-5} [\exp(n - m)\Delta\mu + (\exp(m - n)\,\Delta\mu - 1)^{-1}] C_m R_m \right\} \quad (15)$$

where, because m is always less than n, R_n may be found from previously determined values R_m. The dependence of R_n, B_n, and P_n on T_0 is omitted here and henceforth. The C_m are weight factors evolving from SIMPSON's and COTE's rules.[6] For odd values of n, $C_1 = \frac{4}{15} \ln 2$, $C_2 = \frac{2}{15} \ln 2$, $C_3 = \frac{4}{15} \ln 2$, $C_4 = \frac{2}{15} \ln 2$, $C_5 = \frac{4}{15} \ln 2$, etc. For even values of n, $C_1 = \frac{9}{40} \ln 2$, $C_2 = \frac{9}{40} \ln 2$, $C_3 = \frac{17}{120} \ln 2$, $C_4 = \frac{4}{15} \ln 2$, $C_5 = \frac{2}{15} \ln 2$, $C_6 = \frac{4}{15} \ln 2$, etc. That is, SIMPSON's rule was used for odd values of n and even values of n where $m > 3$. COTE's rule was applied for even values of n where $m = 1$, 2, and 3.

The portion of equation (15) in brackets may be designated as K_{n-m} and equation (15) becomes

$$R_n = B_n^{-1} \left\{ P_n + \sum_{m=0}^{m=n-5} K_{n-m} C_m R_m \right\} \quad (16)$$

where $K_{n-m} = 0$ for $(n - m) = 1, 2, 3, 4, 5$.

The zero values of K_{n-m} correspond to collisions of primary electrons of energy T' where T' does not exceed $2T$, the minimum primary energy for which a secondary electron may be created at energy T. The first few values of R_n are then $R_0 = 0$; $R_1 = B_1^{-1}P_1$; $R_2 = B_2^{-1}P_2$; $R_3 = B_3^{-1}P_3$; $R_4 = B_4^{-1}P_4$; $R_5 = B_5^{-1}P_5$; $R_6 = B_6^{-1}P_6$; $R_7 = B_7^{-1}\{P_7 + K_6C^1R_1\}$; $R_8 = B_8^{-1}\{P_8 + K_7C_1R_1 + K_6C_2R_2\}$, etc.

EXPERIMENT

The electron flux spectrum was measured by a beta spectrometer previously described.[7] This instrument is of the solenoidal, homogeneous field type and accepts radiation from the source in an annular cone whose elements make angles of about 45° with the axis of the solenoid.

The electrons from the spectrometer converge on the detector from a similar annular cone over a range of about one inch along the axis. A special flow type Geiger–Mueller counter was built for the detector. The outer wall consisted of a film of Mylar plastic coated on both sides with aluminum to a total thickness of about 1 mg/cm² and formed into a cylinder $\frac{1}{4}$ in. in diameter and $1\frac{3}{4}$ in. long. This cylinder is fitted over short stubs of tubing through which the center wire leads and the gas tubing entered. Because of the low mass and small size of the counter, its unshielded background was about 4 counts/min. The Mylar window counted electrons of energies down to about 35 keV before cutting off.

The counter was operated at 40 cm pressure with

Tracerlab "Geiger" gas flowing through it. The film would withstand 30 lb/in² differential pressure before rupturing at the seam. Extreme cleanliness was necessary in preparing the wire to avoid spurious counts.

The source was machined of pieces of 50 per cent phosphorus and 50 per cent bakelite used for beta-ray plaques,[8] and irradiated for 3 weeks in the LITR reactor ar Oak Ridge. In order to sample the flux of electrons in the medium, the source was made in the form of a hollow cavity with a small hole to sample the electrons crossing the cavity. The geometrical considerations involved are similar to those encountered in the design of a black body cavity for the measurement of electromagnetic flux. The source, as shown in Fig. 2, consists of three cylindrical discs: a solid back, a center disc with a hole along the axis to form the cavity, and a front piece with a conical aperture to close the cavity and transmit an annular cone of electrons from the cavity to the beta spectrometer. The front disc was made in two pieces, a female cone with a small hole at the apex leading into the cavity, and a male cone mounted on a spider to leave an annular space between the cones.

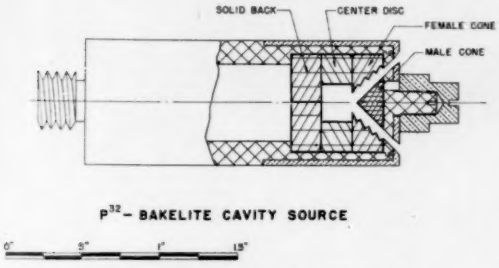


FIG. 2. P^{32} bakelite cavity source.

In the first design, the angle between the axis and an element of each cone was 45° and the cones had smooth surfaces. In this case 70 per cent of the radiation received by the spectrometer came from the cones. In the second, used for the results of Fig. 3, the angle of the female cone was cut back to 40°, and the edges were grooved as shown. This design reduced considerably the number of electrons emerging from the female cone and passing through the spectrometer. By this construction radiation received from the cones amounted to 40 per cent of the total.

RESULTS

The electron flux emerging from the cavity is given by the difference at each energy between the two spectra of Fig. 3. This difference is plotted in Fig. 4 along with the Spencer–Fano–Attix theoretical spectrum. Excellent agreement is seen above 0.35 MeV whereas the experimental spectrum, corrected for transmission loss in

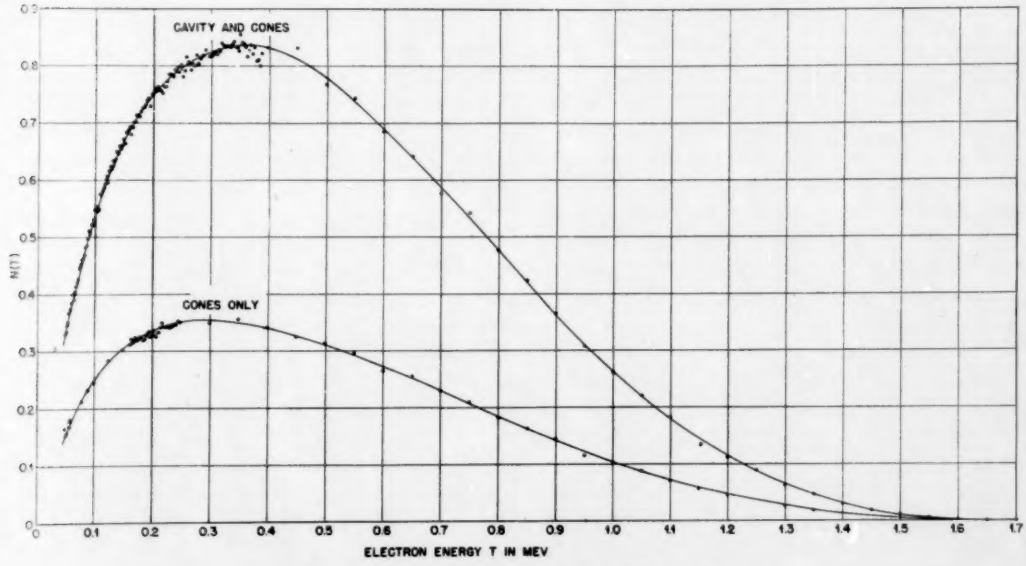


FIG. 3. Electron spectra from P^{32} in bakelite.

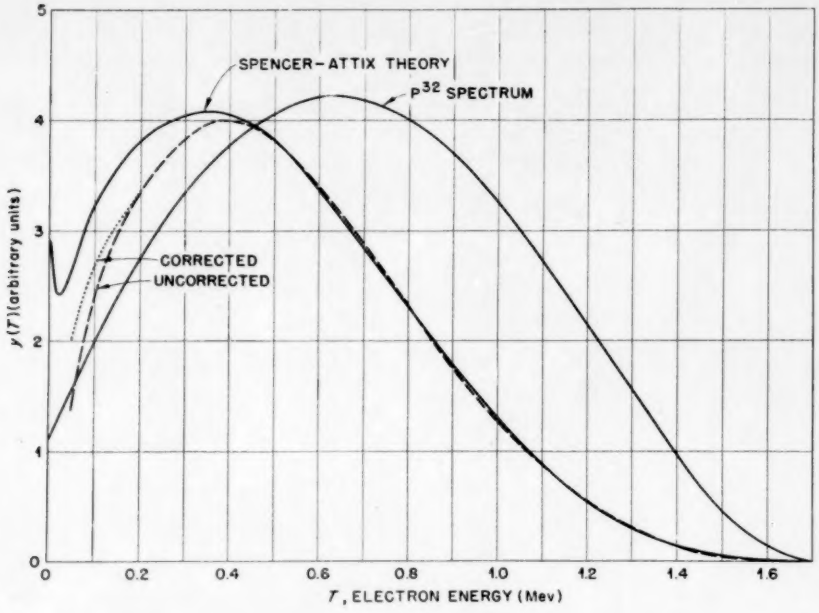


FIG. 4. Spectral distribution of electrons from P^{32} absorbed in bakelite. Curve labeled "P^{32} spectrum" is the unabsorbed nuclear P^{32} spectrum and is shown for purposes of comparison. Solid line is the Spencer–Attix theory and broken line is the experimental energy distribution with and without corrections for spectrometer transmission at low energy.

the beta spectrometer due to counter window absorption, lies somewhat below the theoretical curve between this energy and 0.05 MeV. The discrepancy is probably attributable, in part at least, to uncertainties in the transmission correction mentioned above and to the approximations made in the evaluation of the theory as discussed by SPENCER and ATTIX. In particular the approximation to the integrated MÖLLER formula used in equation (8) and thereafter apparently increases the theoretical flux several per cent at 50 keV.

The nuclear beta spectrum from P^{32} is shown also in Fig. 4. The difference in shape is striking and illustrates the necessity for a careful evaluation of the spectral distribution of electron flux in a medium when dosimeter calibration by immersion in a radioactive medium is performed. Such a calibration has been used recently for the comparison of various beta dosimeters and is discussed in the literature.[9,10]

The theoretical flux spectrum may be seen to increase greatly at the lower energies due to the presence of secondary electrons which are liberated as a result of ionizing collisions by the primary beta rays. As an example, calculations made along the lines indicated above show that the flux is twenty times higher at 119 eV than at 0.322 MeV. It is probable that the flux increases still more as the energy approaches zero although no theoretical evaluation is possible at this time due to lack of any experimental or theoretical information on the stopping power and atomic ionization cross-section at these energies. Because these low energy electrons may play a significant role in radiation damage to biological materials, it would seem highly desirable to study flux spectra at energies approaching zero.[11] Secondary emission experiments,[12] for example, have shown an electron energy distribution having a maximum at around 5 eV.

Acknowledgments—The authors wish to thank R. M. JOHNSON for his aid in constructing the Geiger counter used in this work, and O. SISMAN for the use of his irradiation facility in the Oak Ridge LITR. We are indebted also to R. L. CLARK for his supervision of the source material after irradiation and to L. V. SPENCER for suggesting the calculation procedure which yielded the theoretical flux distribution.

REFERENCES

1. J. R. GREENING, *Brit. J. Radiol.* **27,** 163 (1954).
2. L. V. SPENCER and U. FANO, *Phys. Rev.* **93,** 1172 (1954).
3. L. V. SPENCER and F. H. ATTIX, *Rad. Res.* **3,** 239 (1955).
4. C. MÖLLER, *Ann. Physik.* **14,** 531 (1932).
5. H. A. BETHE, *Hd. Physik.* **24,** 273 (1933).
6. H. and B. S. JEFFREYS, *Methods of Mathematical Physics* p. 287. Cambridge University Press (1950).
7. R. D. BIRKHOFF, A. W. SMITH, H. H. HUBBELL, JR. and J. S. CHEKA, *Rev. Sci. Instrum.* **26,** 959 (1955).
8. O. M. BIZZELL, W. T. BURNETT, JR., P. C. TOMPKINS and L. WISH, *Nucleonics* **8,** No. 4, p. 17 (1951).
9. H. H. HUBBELL, JR., R. M. JOHNSON and R. D. BIRKHOFF, *Nucleonics* **15,** No. 2, 85–89 (1957); *Radiology* **69,** 268 (1957).
10. *Pocket Ion Chambers for Beta Radiation Dose.* Oak Ridge National Laboratory Report No. 2158 (1956).
11. R. L. PLATZMAN, *Rad. Res.* **2,** 1 (1955).
12. K. G. MCKAY, *Advances in Electronics* Vol. 1. Academic Press, New York (1948).

Health Physics Pergamon Press 1958 Vol. 1. pp. 34–38

W FOR HEAVY PARTICLES

LEWIS W. COCHRAN
University of Kentucky, Lexington, Kentucky

(*Received* 22 *November* 1957)

Abstract—The values of *W*, the average energy expended per ion pair produced in a gas, are listed for many pure gases. The change of *W* with small amounts of contaminant gases, the determination of an effective *W* for gas mixtures, and the variation of *W* with velocity for very low velocity particles are considered. Values of *W* for low-energy, high-mass recoil atoms are compared with the values obtained with alpha particles in the same gases.

INTRODUCTION

In radiation dosimetry some of the practical units of measurement are defined in terms of the energy lost by the radiation per gram of tissue. The most commonly used unit is the roentgen equivalent physical (rep), which is defined as that amount of radiation which is absorbed at the rate of 95 ergs of energy per gram of tissue. In practice, the measurements are usually made with a Bragg–Gray cavity containing an appropriate gas. To interpret such measurements quantitatively, whether the cavity is used as a proportional counter or as an ionization chamber, one must know to good accuracy the energy lost by the ionizing radiation for each ion pair produced in the gas. This quantity, designated as *W*, is defined as the average energy in electron volts which is lost by the ionizing particle per ion pair produced in the gas. The value of *W* has been investigated for many years in a variety of gases and gas mixtures using electrons, protons, alpha particles, heavy ions, and fission fragments as the ionizing radiations. There is as yet no really adequate theory for *W* and there are sufficient discrepancies in the experimental results to make impossible any precise generalizations. Experimental results show appreciably lower values for *W* for beta particles than for heavier particles, except in hydrogen and the noble gases. For protons and particles of greater mass it appears quite probable that the value of *W* is independent of the mass and charge of the ionizing particle. For heavy particles of high velocity the experimental results indicate that the value is probably independent of the velocity; for the very low velocities the value of *W* appears to increase with decreasing velocity in the gases investigated. Each gas has its own characteristic value of *W*. In some instances the experimental value is quite sensitive to small amounts of impurities in the gas, and many of the early measurements were hampered by impure gases as well as instrumental difficulties. The earlier work has been surveyed and discussed in some detail by Gray.[1] Some of the more recent measurements involving protons and heavier particles will be summarized in this present paper.

MEASUREMENTS WITH ALPHA PARTICLES

Most recent measurements of *W* for heavy particles have utilized alpha particles from radioactive sources, and ionization chambers of such dimensions and at such gas pressures that the total energy of the alpha particle is lost in the active volume of the ionization chamber. The detecting system used either collected the total ionization produced—electrons, positive ions, and negative ions—or else correction was made for the heavier ions not collected.

Values obtained for pure gases by Bortner and Hurst,[2] by Jesse and Sadauskis,[3] by Valentine and Curran,[4] and by Haeberli, Huber and Baldinger,[5] are summarized in Table 1. It will be noted that with the exception of helium, the values of *W* for any gas are in excellent agreement. The probable errors shown

Table 1. *Values of W in* eV/*ion pair for alpha particles*

Gas	Bortner and Hurst	Jesse and Sadauskis	Valentine and Curran	Haeberli, Huber and Baldinger
Argon	26.4 ± 0.3	26.4	25.9	26.25
Helium	46.0 ± 0.5	42.7	31.7	30.86
Neon	—	36.8	—	—
Krypton	—	24.1	—	—
Xenon	—	21.9	—	—
Hydrogen	37.0 ± 0.4	36.3	37.0	—
Nitrogen	36.3 ± 0.4	36.6	36.0	36.3
Carbon dioxide	34.3 ± 0.3	34.5	—	33.5
Boron trifluoride	36.0 ± 0.4	—	—	—
Oxygen	32.2 ± 0.3	32.5	32.2	32.17
Air	35.0 ± 0.3	35.5	35.2	—
Sulfur hexafluoride	35.7 ± 0.4	—	—	—
Freon-12	29.5 ± 0.3	—	—	—
Acetone	28.5 ± 0.3	—	—	—
Acetylene	27.7 ± 0.3	27.5	—	—
Benzene	27.5 ± 0.3	—	—	—
Butane	26.4 ± 0.3	—	—	—
Ethane	26.7 ± 0.3	26.6	—	—
Ethylene	28.0 ± 0.3	28.0	—	—
Isobutane	26.1 ± 0.3	—	—	—
Methane	29.4 ± 0.3	29.2	29.0	—
Methyl iodide	24.8 ± 0.2	—	—	—
Propane	26.3 ± 0.3	—	—	—

in this table are due mainly to uncertainty in the disintegration rate of the alpha source alone; however, this is the principal uncertainty in the experiment.

The importance of having very pure gases and the effect of even trace amounts of impurities have been shown best by the measurements on helium, neon and argon. Early measurements on helium gave a *W*-value of about 30 eV/ion pair. Although it has not been possible to calculate *W*-values with precision, Williams[6] in 1932 did arrive at an expected range of values for hydrogen consistent with experimental results, while his expected value of *W* for helium was found to be between 55 and 80 eV/ion pair. As a possible explanation of the wide discrepancy between his calculated range of values and the experimental value of about 30 eV/ion pair, Williams suggested that impurities interacting with the excited and ionized helium atoms and ions might themselves become ionized, leading to a low experimental value of *W*. Recent measurements by Jesse and Sadauskis[7] and by Bortner and Hurst[8] give experimental values of 42.7 and 46.0 eV/ion pair respectively. Careful gas purification and the addition of small amounts of contaminant gases have shown that the previous low values of *W* were due to impurities. By way of further explanation, Jesse and Sadauskis suggest that the previous low reported values for helium result from the transfer of energy from the well-known metastable state in helium at 19.7 V to the contaminating gas, with the production of extra ionization in gases which have an ionization potential less that the energy of the metastable state. Theoretical treatments by Erskine[9,10] are consistent with this explanation. Similar results have been found by Jesse and Sadauskis[7] in neon.

Mixtures of various contaminant gases in argon have been investigated in detail by MELTON, HURST and BORTNER.[11] It was found that there is a particular concentration for each contaminant gas which gives the maximum decrease in the value of W for the mixture. It was also found that while contaminant gases with ionization potentials less than the metastable potential of 11.5 V in argon result in a lowering of the W-value of the mixtures, consistent with the results found for helium and neon, several contaminant gases with ionization potentials greater than 11.5 V also result in some lowering of the W-value. In explanation, it has been suggested that the lower W-value resulting from the increase in ionization is due to a transfer of energy from an excited level in argon to the impurity gas.

Certain gas mixtures are sometimes required for proper operation of counters, and it is frequently desirable to be able to compute an effective W-value for a gas mixture composed of constituent gases with known W-values. In general investigations of gas mixtures BORTNER and HURST have found that an effective value W_m for gas mixtures can be calculated using the empirical equation

$$\frac{1}{W_m} = \left(\frac{1}{W_1} - \frac{1}{W_2}\right) Z + \frac{1}{W_2} \qquad (1)$$

where $$Z = P_1/(P_1 + aP_2) \qquad (2)$$

and P_1 is the partial pressure of the gas having a value of W_1, and P_2 is the partial pressure of the gas having a value of W_2. Values of a, an empirical constant for each gas mixture, are given in Table 2.

Table 2. Values of a for some gas mixtures

N_2–H_2	0.28	He–N_2	8.47
N_2–A	0.53	He–CH_4	0.68
N_2–O_2	1.06	A–H_2	0.56
He–A	0.75	H_2–CH_4	4.03
He–H_2	3.55		

VARIATION IN W WITH PARTICLE VELOCITY

It is to be emphasized that the measurements of W previously discussed were performed in such a way that the ionization collected was the total ionization produced in the gas by the complete loss of energy by the particle. This means that the value of W so obtained is truly an average value for that energy particle, and any variations in W with velocity might not be readily detected if these variations were limited to the very low velocity portion of the path of the particle. This problem has been given some study in recent years.

Measurements by JESSE, FORSTAT and SADAUSKIS[12] using alpha particles from radioactive sources have found no variation in W for argon with particle energies between 5.3 and 8.8 MeV. Using data from nuclear reaction measurements, these authors find the same constancy for alpha energies down to 0.8 MeV. However, direct measurements of the total ionization produced in air and in argon by JESSE and SADAUSKIS[13] show that the ratio of the ionization produced due to samarium and polonium alpha particles in argon is 5 per cent higher than the same ratio in air. Inclusion of other alpha particle data from radioactive sources and nuclear reactions indicates an increase of W with decreasing particle energy in air of as much as 16 per cent between 5 MeV and zero. Direct investigation of the ionization produced in air by alpha particles by KIMURA *et al.*,[14] has shown that much of this apparent variation of W with energy in air is probably due to columnar recombination and lack of voltage saturation in the ionization chamber.

The ionization produced in argon by protons from an accelerator has been investigated by LOWRY and MILLER[15] in the energy region 25–250 keV. A total increase of about 5 per cent in the value of W was found for this decrease in energy. An extrapolation of the ionization vs. energy curve to zero ionization led to an intercept of 1.4 keV; this intercept is called the ionization defect.

Proportional counter measurements in hydrogen and methane, utilizing knock-on protons resulting from bombardment with monoenergetic neutrons, have been made by ALLEN and FERGUSON.[16] These results show a linear relationship between ionization and energy down to 10 keV for hydrogen; the curve for

methane between 100 and 1000 keV was not linear through the origin, having an intercept for zero ionization of between 20 and 40 keV.

VALUES FOR HEAVIER PARTICLES

The ionization produced in gases by fission fragments has been studied by many investigators. The work of HERWIG and MILLER[17] and of SCHMITT and LEACHMAN[18] might be cited as particularly pertinent in this discussion. In these experiments, the ionization produced in the various gases and gas mixtures is measured for different fragment energies, and the data fitted to an assumed linear relationship between ionization and energy. These linear plots do not extrapolate to zero energy for zero ionization. The same W-value as for alpha particles in the gas studied is assumed, and the resulting intercepts lead to ionization defects of about 6 and 7 MeV respectively for the lighter and heavier groups of fission fragments.

The energy associated with the recoil of heavy atoms following radioactive decay is not large enough to be a major factor in dosimetry considerations. However, the use of these heavy ions resulting from the recoil from radioactive decay offers a convenient means of studying the value of W for large-mass, low-velocity particles. JESSE and SADAUSKIS[19] have measured the value of W in argon and in helium using the recoil atom Pb^{206} resulting from the alpha decay of Po^{210}. Their technique involved a hemispherical ionization chamber with the source on a flat collecting electrode at the geometrical center. The ionization current was measured as a function of the pressure of the gas in the chamber and extrapolation of the straight portion of this curve gave an intercept from which the W-value could be calculated. Their results gave values of 4.5 and 4.1 times the values obtained with alpha particles in argon and helium respectively.

Earlier, a proportional counter in a coincidence arrangement was used by MADSEN[20] in an investigation of the ionization produced by the recoil atoms from the alpha decay of Po^{210}, ThC(Bi^{212}), and ThC′(Po^{212}) in a mixture of 95 per cent argon and 5 per cent air. These measurements showed the W-values to be from 3.4 to 4.5 times the value obtained for alpha particles in the same gas mixture and to increase with decreasing particle energy.

A somewhat similar method has been employed recently by STONE and COCHRAN.[21] This experiment utilized a ThC–ThC′ source and extended the measurements to a variety of gases for which dE/dx (energy loss per unit path length) data were available, and which were suitable for proportional counter operation. The W-values in electron volts per ion pair for the recoil atoms from the alpha decay of ThC and ThC′ are shown in Table 3 for the several gases considered. These values are significantly lower than the values of MADSEN and of JESSE and SADAUSKIS, the ratio of W for the recoil atoms to W for alpha particles ranging from 1.8 to 3.8 while the ratios from the earlier work ranged from 3.4 to 4.5.

Table 3. *W values* (eV/*ion pair*) *for particles from alpha decay of* ThC *and* ThC′

Gas	W_α	ThC		ThC′	
		W_{recoil}	$\frac{W_{recoil}}{W_\alpha}$	W_{recoil}	$\frac{W_{recoil}}{W_\alpha}$
Carbon dioxide	34.3	102	2.98	99	2.88
Cyclopropane	25.9	99	3.68	92	3.55
Ethylene	28	104	3.72	90	3.21
Helium*	30	60	2.00	55	1.83
Hydrogen	37	81	2.19	68	1.84
Methane	29.4	111	3.78	96	3.26

* Impure helium values.

SUMMARY

From the standpoint of the needs of radiation dosimetry, W-values for the gases commonly used in ionization chambers and proportional counters are now known to good accuracy. For those instances where it is desirable to use gas mixtures, techniques and data are available for the proper interpretation of the results. The value of W is accepted as essentially constant for protons, alpha particles and fission fragments of high velocity in any particular gas. It seems probable that there is some increase in W for all

gases at very low particle velocities, this increase being greater in some gases than in others. Although the contribution of the low velocity particle and of the heavy recoil or knock-on atom to the total radiation dose is usually a very small part of the total dose, it would seem that further investigation of these effects would be desirable.

REFERENCES

1. L. H. Gray, *Proc. Camb. Phil. Soc.* **40,** 72 (1944).
2. T. E. Bortner and G. S. Hurst, *Phys. Rev.* **93,** 1236 (1954).
3. W. P. Jesse and J. Sadauskis, *Phys. Rev.* **90,** 1120 (1953).
4. J. M. Valentine and S. C. Curran, *Phil. Mag.* **43,** 964 (1952).
5. W. Haeberli, P. Huber and E. Baldinger, *Helv. Phys. Acta* **25,** 467 (1952).
6. E. J. Williams, *Proc. Roy. Soc.* A **35,** 108 (1932).
7. W. P. Jesse and J. Sadauskis, *Phys. Rev.* **88,** 417 (1952).
8. T. E. Bortner and G. S. Hurst, *Phys. Rev.* **90,** 160 (1953).
9. G. A. Erskine, *Proc. Roy. Soc.* A **224**, 362 (1954).
10. G. A. Erskine, *Proc. Phys. Soc.* A **67,** 640 (1954).
11. C. E. Melton, G. S. Hurst and T. E. Bortner, *Phys. Rev.* **96,** 643 (1954).
12. W. P. Jesse, H. Forstat and J. Sadauskis, *Phys. Rev.* **77,** 782 (1950).
13. W. P. Jesse and J. Sadauskis, *Phys. Rev.* **75,** 1110; **76,** 163 (1949).
14. K. Kimura, R. Ishawari, K. Yuasa, S. Yamashita, K. Miyake and S. Kimina, *Phys. Soc. Japan. J.* **7,** 111 (1952).
15. R. Lowry and G. H. Miller, *USAEC Report* ISC 655 (1955).
16. W. D. Allen and A. T. G. Ferguson, *Harwell Report* AERE NP/R 1720 (1955).
17. L. O. Herwig and G. H. Miller, *Phys. Rev.* **95,** 413 (1954).
18. H. W. Schmitt and R. B. Leachman, *Phys. Rev.* **102,** 183 (1956).
19. W. P. Jesse and J. Sadauskis, *Phys. Rev.* **102,** 389 (1956).
20. B. S. Madsen, *Kgl. Danske Vidensk. Selsk., Mat.-fys. Medd.* **23,** No. 8, (1945).
21. W. G. Stone and L. W. Cochran, *Phys. Rev.* **107,** 702 (1957).

Health Physics Pergamon Press 1958. Vol 1. pp. 39–45

AN APPARATUS FOR MEASURING ELECTRON ATTACHMENT: RESULTS FOR OXYGEN IN ARGON

T. E. BORTNER and G. S. HURST*
Oak Ridge National Laboratory, Operated by Union Carbide Nuclear Company for the Atomic Energy Commission, Oak Ridge, Tennessee

(*Received* 19 *December* 1957)

Abstract—The time variation of the electrostatic potential of a plane electrode ionization chamber, due to the motion of electrons in the uniform field, depends on the drift velocity of electrons and the attachment of the electrons to the gas in the chamber. If the pulse is examined with a linear pulse amplifier having suitable integration and differentiation time constants, the pulse height will be a simple analytical function of the attachment coefficient and the electron drift velocity. This information has been used to develop a pulse method of measuring electron attachment. In one section of a plane electrode ionization chamber the pulse height due to the motion of electrons, produced by alpha particle ionization, is measured. Drift velocity of electrons is measured in another part of the same ionization chamber. These two measurements furnish data from which the attachment coefficient is calculated. Results for trace amounts of oxygen in argon will be presented.

1. INTRODUCTION

A MAJOR difficulty in the use of ionization chambers, proportional counters, and Geiger–Mueller tubes is the problem of electron attachment. Attachment is defined here as a process in which free electrons react with an atom or molecule to form a heavy negative ion. In the case of pulse ionization chambers, an attached electron becomes less effective in producing a pulse, whereas in a proportional or G–M counter, the attached electron completely escapes detection, unless the less likely effect of electron detachment takes place. It is generally assumed[1] that water-vapor and oxygen will be the most common offenders to ionization detection devices, yet at least in the case of oxygen there are no data which will even indicate the order of magnitude of the severity of the effect of attachment on, for example, the diminution of pulse height in a plane geometry ionization chamber. This comes about because of two difficulties: (1) the evidence for[2] or against[3] the pressure dependence of the attachment coefficient is rather vague, and (2) most of the studies have been done with pure oxygen. Thus the data are not applicable to the case where oxygen is a trace amount in some other principal gas, without more knowledge than is now available on the agitation energy of electrons.[4]

This paper will report a method of measuring electron attachment which gives the data already in the practical form for predicting the influence of attachment on ionization devices. Further interpretation of the results leads to a better understanding of the basic mechanisms of the attaching process.

2. METHOD

We chose the effect of electron attachment on the pulse height in a plane geometry ionization chamber as the basis of a method of measuring attachment coefficients. Suppose N_0 electrons are formed at a distance d (Fig. 1) from a positively charged electrode. Because of attachment, at a distance x from the point of formation only N of the electrons are free (i.e. not attached). The number dN attached in a distance dx is given by:

$$dN = -\alpha N f_1 P \, dx \qquad (1)$$

which defines α as the probability of attachment per

* This material will be submitted by G. S. HURST as a part of a dissertation for the degree of Doctor of Philosophy at the University of Tennessee.

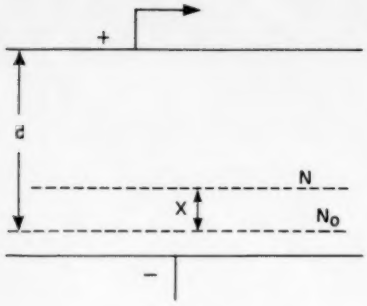


FIG. 1.

centimeter of travel in the field direction and per millimeter of partial pressure (f_1P) of the attaching gas. Let $P = f_1P + f_2P$, where f_2P is the partial pressure of a non-attaching gas. In this work, $f_1P \ll f_2P$; thus E/P refers approximately to the "reduced electric field"—volts per centimeter and per millimeter Hg of the non-attaching gas. Integration of equation (1) yields:

$$N(x) = N_0 \exp(-\alpha f_1 P x) \tag{2}$$

for the number of free electrons at distance x. The change in electrode potential dV due to the work done by the field in moving $N(x)$ electrons a distance dx is:

$$dV = N(x)\frac{V_0}{d}\,dx \tag{3}$$

where V_0 is the change in potential due to a single electron moving the entire distance d. Integration of equation (3) using equation (2) for $N(x)$ shows that the change of potential $V(x')$ after the electrons have moved a distance x' is

$$V(x') = \frac{N_0 V_0}{\alpha f_1 P d}[1 - \exp(-\alpha f_1 P x')]. \tag{4}$$

The time variation of the electrode potential is found by replacing x' by td/τ_0 where τ_0 is the "collection time" and is related to the electron drift velocity W by $\tau_0 = d/W$. Thus, after replacing N_0V_0 by A and $\alpha f_1 P d$ by f, equation (4) gives

$$V(t) = \frac{A}{f}(1 - \exp(-ft/\tau_0)\,. \tag{5}$$

When α (hence f) approaches zero, equation (5) reduces to the familiar expression $V(t) = At/\tau_0$ for $0 \geq t \leq \tau_0$, which represents the pulse profile in a plane ionization chamber for the non-attachment case.

Let us now assume that the pulse represented by equation (5) is observed with a linear pulse amplifier having a response to a step function

$$V'(t) = \frac{t}{t_1}\exp(-t/t_1) \tag{6}$$

where t_1 is the amplifier differentiating and integrating time constant (assumed equal). The output of the pulse amplifier will then be given by

$$V(\tau) = \int_0^{\tau_0} \frac{dV(t)}{dt}\,V'(\tau - t)\,dt \tag{7}$$

for $\tau > \tau_0$.

On integration, equation (7) becomes

$$V(\tau) = \frac{A}{(\tau_0 - t_1 f)}\left[(\exp\{(\tau_0 - t_1 f)/t_1\} - 1)\tau - \exp\{(\tau_0 - t_1 f)/t_1\}\left(\tau_0 - \frac{t_1\tau_0}{\tau_0 - t_1 f}\right) - \frac{t_1\tau_0}{\tau_0 - t_1 f}\right]\exp(-\tau/t_1). \tag{8}$$

This expression has a maximum when $\tau = \tau' \geq \tau_0$ where

$$\tau' = \frac{\exp\{(\tau_0 - t_1 f)/t_1\}\left[t_1 + \tau_0 - \frac{t_1\tau_0}{\tau_0 - t_1 f}\right]}{[\exp\{(\tau_0 - t_1 f)/t_1\} - 1]} + \frac{t_1\left[\frac{\tau_0}{\tau_0 - t_1 f} - 1\right]}{[\exp\{(\tau_0 - t_1 f)/t_1\} - 1]}. \tag{9}$$

The pulse height $V(\tau')$ is determined by insertion of τ' for τ into equation (8).

The pulse height is plotted in Fig. 2 as a function of

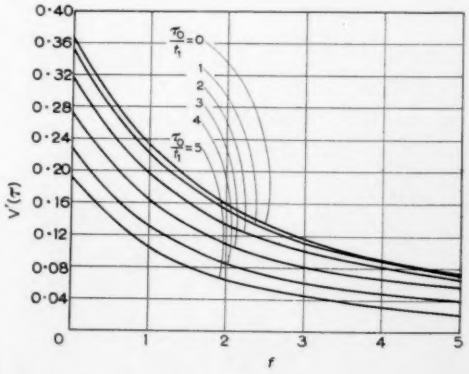


FIG. 2.

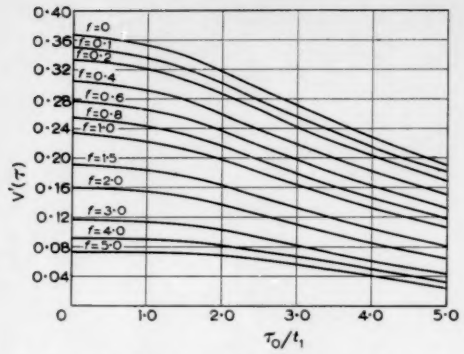


FIG. 3. Calculated dependence of pulse height on τ_0/t_1 for various values of f.

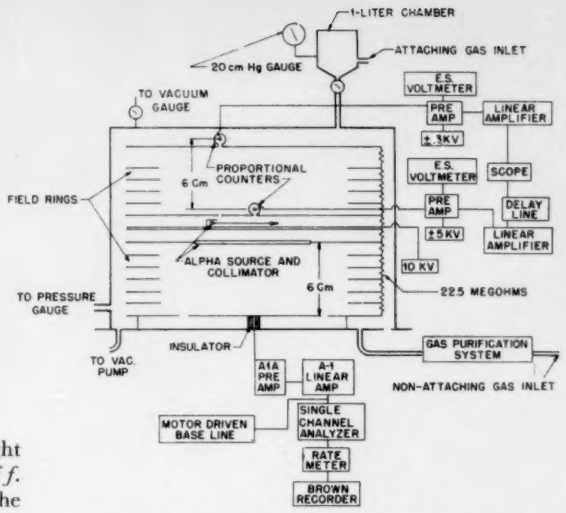


FIG. 4. Electron attachment apparatus.

f for various values of τ_0/t_1. In Fig. 3 the pulse height is plotted as a function of τ_0/t_1 for various values of f. These curves may then be used to calculate the attachment coefficient from the experimentally measured quantities $V(\tau')$ and τ_0.

We have tacitly assumed that the amplifier constant t_1 is such that the response to the slowly moving negative ions can be neglected. This is the case when t_1 is of the order of τ_0, since, in general, the drift velocity of heavy ions is about one-thousandth that of the drift velocity of electrons.

3. APPARATUS

Since data on electron drift velocity is inadequate, especially for mixtures of gases, it is important to measure the drift velocity as well as pulse height. Furthermore, it is known that in many gases the drift velocity is extremely sensitive to contamination. The apparatus shown in Fig. 4 permits measurement of both quantities in the same enclosure; thus even if a non-attaching contamination should affect the drift velocity in the gas mixture under study, the calculated attachment coefficient would be unaffected. The main vessel is made of stainless steel, and is sealed by teflon 0 rings. To improve uniformity of the electric field, rings are spaced 1 cm apart with fluorothene insulators. These are held at appropriate potentials with a voltage divider system using four 5.6 MΩ (1 W Allen Bradley) resistors per stage. In the upper part of the chamber the drift velocity of electrons is measured with two proportional counters, as already described by BORTNER, HURST and STONE.[5] The pulse height resulting from the electrons produced by collimated (in a plane parallel to the collecting electrode) Pu^{239} alpha particles is measured in the lower chamber. The collector to source distance d is normally 6.0 cm, although the apparatus is designed to permit various separations.

A linear pulse amplifier of the JORDAN–BELL[6] type was modified so that the long time constant position would give a pulse shape like that described in equation (6) with $t_1 = 15\mu$ sec. The response of the particular amplifier used in this experiment to a step function pulse is shown in Fig. 5. With $t_1 = 15$ μsec, the agreement with the idealized response [equation (6)] is good up to times of 50 μsec. Deviation from the idealized response will not, however, affect the pulse height provided the amplifier output pulse peaks in a time τ' less than 50 μsec. From equation (9) it can be seen that

$$\frac{d\tau'}{df} = \frac{-(e^{\mu}-1)^2\,\tau_0/\mu^2 + \tau_0 e^{\mu}}{(e^{\mu}-1)^2} \qquad (10)$$

where $\mu = (\tau_0 - t_1 f)/t_1$. Now $d\tau'/df < 0$ for all positive f; therefore, τ' will be greatest for $f = 0$. In this case

$$\tau' = \frac{\tau_0 \exp(\tau_0/t_1)}{(e^{\tau_0/t_1}-1)} \leqq (\tau_0 + t_1)\,. \qquad (11)$$

We require then that the experiment be

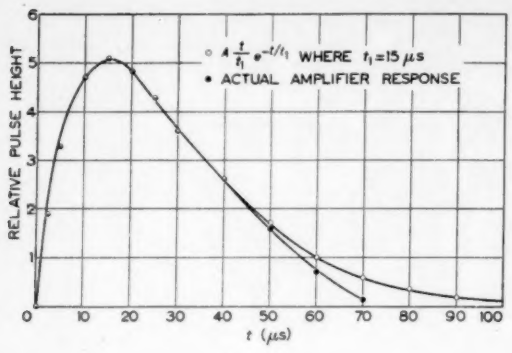


FIG. 5.

performed such that $(\tau_0 + t_1)$ is always less than 50 μsec.

Pulse height is measured with a single channel analyzer having a motor-driven base line so that the "window" scans uniformly in time. A display of the analyzer output on a count rate recorder affords a convenient method of pulse height measurement.

A gas purification system is of necessity "tailored" to the particular gas to be purified and to the gases which are to be removed. A drying unit is necessary regardless of the gas to be purified, since water-vapor may be present, and is extremely effective in reducing the pulse height due to electron attachment Either P_2O_5-Drierite or anhydrone is used in two inconel traps (in series) 18 in. long and 1.5 in. in diameter. When feasible, the gases are further purified by distillation (or sublimation) with a simple trap surrounded with liquid N_2. In other cases, it is helpful to absorb virtually all gases in activated coconut charcoal[7] held at liquid N_2 temperature. The desired gas may then be released by increasing the temperature of the charcoal.

4. RESULTS

(a) *Experimental check of dependence of pulse height on τ_0/t_1 using pure* CO_2

The dependence of pulse height on E/P (V/cm/mm Hg) was studied in pure CO_2. In CO_2 the electron drift velocity[5] varies nearly linearly with E/P going from a very low value of 0.1 cm/μsec at $E/P = 0.2$ to 1.0 cm/μsec at $E/P = 2.0$. For a 6.0 cm source to plate separation corresponding values of τ_0 are 60 and 6 μsec, respectively. Pulse height measured with an amplifier having a value $t_1 = 15$ μsec would vary considerably with E/P in CO_2 (see Fig. 3 for $f = 0$).

Very careful purification of "bone dry" CO_2 is required. If CO_2 is admitted directly into the chamber, no pulse can be found at any value of E/P (i.e. the pulse height is less than amplifier noise pulses). When the CO_2 is passed through the anhydrone drying traps only, the pulse height is about half the value found when the CO_2 is passed through anhyrdone and then further purified with a cold trap. The cold trap is filled with copper turnings and is surrounded with liquid nitrogen. At this temperature sublimation takes place and oxygen can be removed by pumping. When the temperature is increased very pure CO_2 goes into the attachment chamber, giving a highly reproducible pulse height vs. E/P. This curve is shown in Fig. 6 where a comparison is made with the calculated pulse height curve, obtained by normalization of the $f = 0$ curve in Fig. 3 and using the values in (5) for drift velocity in CO_2.

(b) *Attachment of electrons in argon–oxygen mixtures*

As shown in Section 2, the attachment coefficient α may be determined by measuring the drift velocity and pulse height due to the motion of free electrons in a plane geometry ionization chamber. In this section we will show results for argon–oxygen mixtures.

Argon was purified by first removing the oxygen with a Baker Deoxo trap, next removing water-vapor with the anhydrone trap, and then removing nitrogen by pumping while holding

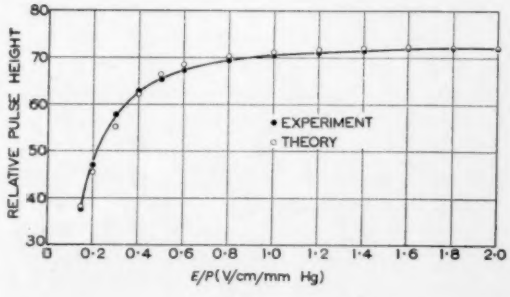


FIG. 6.

argon in the liquid nitrogen trap. The last step was repeated as many times as necessary to yield a pulse height versus E/P curve having the calculated behavior for pure argon.

Drift velocity curves for argon–oxygen mixtures are shown in Fig. 7. The behavior of these curves is rather striking, both the shape and magnitude being strongly affected by the addition of less than 1 part oxygen per 1000 parts argon. The data shown were taken at an argon pressure of 800 mm Hg. Several checks were made by reducing the argon pressure to 600 and 400 mm Hg. The drift velocity at a given E/P and a fixed concentration of oxygen was not affected by total pressure. The increase

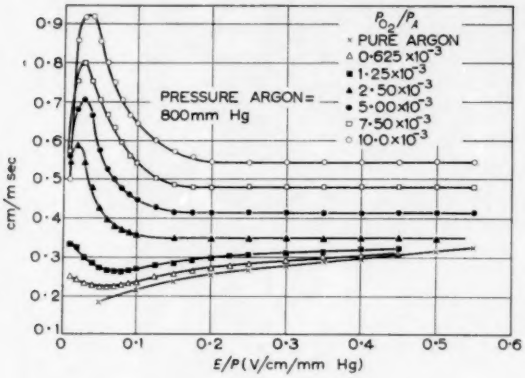


FIG. 7.

in drift velocity with the addition of oxygen implies a decrease in the electron agitation energy, according to the usual qualitative argument.

The pulse height was determined as a function of E/P, total pressure, and concentration of oxygen, as shown in Figs. 8, 9 and 10. In was seen early in the work that the attachment coefficient appeared to depend on pressure; thus data were taken at three pressures of argon (400, 600 and 800 mm Hg) and the pressure of oxygen was varied for each of the argon pressures, but in such a way that sets of fixed fractions $(P_{O_2})/(P_A)$ (called oxygen concentration) could be established. At values of E/P less than 0.3 the pulse height actually increases with the addition of oxygen. This can be accounted for by the increase in drift velocity, but the small values for the attachment

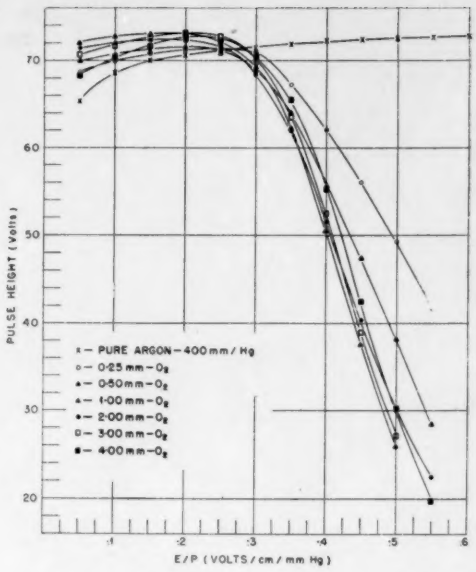


FIG. 8.

coefficients at the low E/P cannot be determined with accuracy since the drift velocity correction on pulse height is so large. A very interesting behavior of practical importance is seen at $E/P = 0.30$ (Fig. 8). At this E/P oxygen (or air) contamination would not appreciably

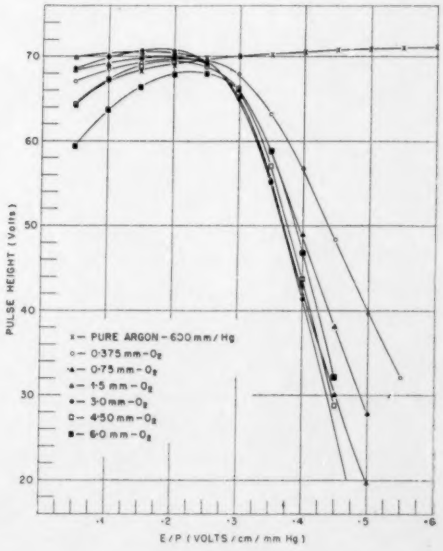


FIG. 9.

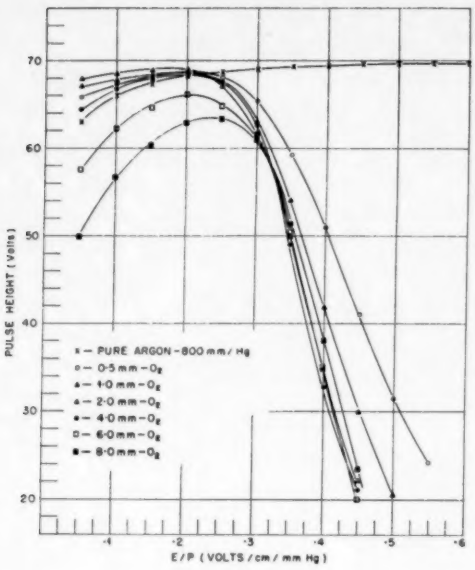


FIG. 10.

decrease the pulse height in an ionization chamber.

At E/P values greater than 0.30 the attachment coefficient can be calculated. Results are shown in Fig. 11 for $E/P = 0.35$, 0.40 and 0.45. Curves for α are plotted as a function of oxygen concentration for these E/P values, and for argon pressures 400, 600 and 800 mm Hg. The attachment coefficient depends strongly on E/P and oxygen concentration, but is independent of argon pressure. This behavior of α can be interpreted without difficulty. The attachment reaction which is involved is $O_2 + e \rightarrow O^- + O$ since in argon at these E/P values the electron energy is great enough to dissociate the O_2 molecule.[8]

BRADBURY's data[3] for the attachment cross-section peaks at about 2 V, and he has explained this process of attachment as being due to inelastic collisions with O_2 in which the electron loses energy and is then captured directly to O_2, forming O_2^-. This process would not be expected to enter into this case where the number of O_2 molecules compared to A atoms is small. Therefore, we may conclude on the basis of the electron energy distribution shown by BARBIERE,[9] that the only process which is likely to lead to electron attachment is the formation of O^- by molecular dissociation.

Fig. 11 shows that as E/P decreases, α decreases and since a decrease in E/P must be associated with a decrease in the mean agitation energy, α (in this region of electron energies) decreases with energy. Electron agitation energy depends on the reduced electric field, E/P, and the nature of the gas. In argon there are no excitation levels less than 10 V, consequently electrons may be accelerated by the field to energies up to 10 V while making only elastic collisions with argon atoms. When O_2 is added to argon, the mean electron energy is decreased because electrons make inelastic collisions with O_2. Therefore, for a fixed argon pressure, α will decrease with increasing values of P_{O_2}/P_A. On the other hand, if P_{O_2}/P_A is fixed the electron agitation energy will not depend on total pressure, hence α should not depend on P_A.

The magnitude of α can best be judged by computing h, the probability of capture per collision with O_2, which is related to α through

$$h = \frac{\alpha}{C}\frac{m}{e}\frac{W^2}{E/P}\left[f_1 + f_2\frac{\lambda_{10}}{\lambda_{20}}\right] \qquad (12)$$

where W is the drift velocity of electrons; λ_{10} and λ_{20} are the mean free path of electrons at 1 mm pressure in O_2 and N_2, respectively; C is a numerical constant which depends on the energy distribution of electrons, but is assumed unity here; f_1 and f_2 are the mole fractions of O_2 and A, respectively; and m and e are the mass and charge of the electron, respectively.

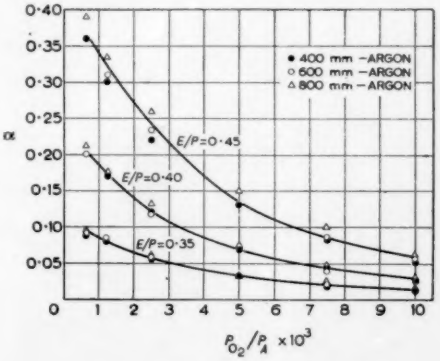


FIG. 11.

If the extrapolated values of α are used for $P_{O_2}/P_A = 0$, and if values of λ_{10} and λ_{20} are taken from HEALEY and REED,[4] h is 1.35×10^{-5}, 0.83×10^{-5}, and 0.35×10^{-5} for the E/P values of 0.45, 0.40, and 0.35, respectively. The magnitudes of these values are well under the 10^{-3} maximum quoted by WILKINSON.[1]

Acknowledgments—The authors are pleased to acknowledge technical assistance by J. A. HARTER and W. G. STONE of the Oak Ridge National Laboratory.

REFERENCES

1. D. H. WILKINSON, *Ionization Chambers and Counters*. Cambridge University Press (1950).
2. H. S. W. MASSEY and E. H. S. BURHOP, *Electronic and Ionic Impact Phenomena*. Oxford University Press (1952).
3. N. E. BRADBURY, *Phys. Rev.* **44,** 883 (1933).
4. R. H. HEALEY and J. W. REED, *The Behavior of Slow Electrons in Gases*. The Wireless Press for Amalgamated Wireless Ltd., Sydney, Australia (1941).
5. T. E. BORTNER, G. S. HURST and W. G. STONE, *Rev. Sci. Instrum.* **28,** 103 (1957).
6. W. H. JORDAN and P. R. BELL, *Rev. Sci. Instrum.* **18,** 10 (1947).
7. DUSHMAN, *Vacuum Techniques*. John Wiley, New York (1949).
8. H. D. HAGSTRUM, *Rev. Mod. Phys.* **23,** 185 (1951).
9. D. BARBIERE, *Phys. Rev.* **84,** 653 (1951).

Health Physics Pergamon Press 1958. Vol. 1. pp. 46–50

DOSIMETRY WITH IONIZATION CHAMBERS EMPLOYING INTERNAL GAS MULTIPLICATION*

HARALD H. ROSSI and M. LUBERT
Radiological Research Laboratory, Columbia University, New York City

(*Received* 26 *November* 1957)

Abstract—In shallow ionization chambers, current densities in the range from 10^{-12} to 10^{-15} A/cm^2 may be increased by factors in excess of 10^4 by the application of high potentials. Reproduction is adequate provided regulated voltage sources are used and gas purity is maintained. The multiplication was found to be independent of current density. The method should be of value in the field of radiation protection, where it would lend itself particularly well to the accurate measurement of beta-ray doses.

When a broad beam of ionizing radiation is normally incident on an interface between media of different composition (such as air and tissue) the pattern of energy absorption may vary rapidly along the beam axis. Thus in the case of beta radiation striking the skin, absorption and scattering result in a depth dose curve that approaches zero value within a few millimeters of tissue. A correct value of the dose at any depth may be obtained by using the Failla extrapolation chamber.[1] In the use of this instrument the current per unit volume of gas is plotted versus electrode spacing. The shape of this curve depends somewhat on geometrical conditions and on the energy of the radiation employed; however, it will be found, in general, that when the spacing is of the order of ¦ mm the current per unit volume does not differ greatly from the extrapolated value. Thus a reasonably accurate result may be obtained from a single measurement employing a very shallow ionization chamber.

When one is dealing with dose rates that are near permissible levels the ionization currents that one may obtain from a shallow chamber are quite small. Thus for a dose rate of 10 mrads/hr the current observed will be of the order of 10^{-15} A/cm^3. Hence, for a chamber of 5 cm diameter and 1 mm depth the total current is about 2×10^{-15} A. While larger currents may be obtained employing larger diameters, the attendant increase of capacity tends to offset and ultimately cancel the advantage, since the voltage rise of the collecting system will tend to become independent of chamber diameter. In addition, small diameter is essential if the beta ray source is small or not uniform. On the other hand, the deep ionization chambers incorporated in survey meters will register dose rates that may be in error by a factor of 10 and more.[2]

Since electronic current amplification represents a difficult problem at these levels, chambers were developed in which gas multiplication could be attained by the deliberate application of voltages that are in excess of those required for saturation. The parallel plate chamber lends itself very well to this purpose since the electric field is uniform and breakdown problems are minimized.

THEORETICAL CONSIDERATIONS

If the electric field is uniform and if multiplication occurs only by electron collisions (Townsend discharge), a number of electrons, n, traversing a thin layer, dx, will be increased to $\alpha n\, dx$, where α is the first Townsend coefficient. If the radiation produces q electrons per unit

* Based upon work performed under AT-30-1 Gen. 70 of the Atomic Energy Commission.

volume, the total change in electron density is given by:

$$dn = \alpha n\, dx + q\, dx \quad . \quad . \quad . \quad .$$

The solution of this equation is:

$$n = \frac{q}{\alpha}(e^{\alpha x} - 1) \quad . \quad . \quad . \quad .$$

If ε is the electronic charge, the current density in the chamber, J, is given by:

$$J = \frac{q\varepsilon}{\alpha}(e^{\alpha x} - 1) \quad . \quad . \quad . \quad .$$

In the absence of multiplication the saturation current density, J_0, is given by:

$$J_0 = q\varepsilon d$$

hence for a separation, d, the observed current density is

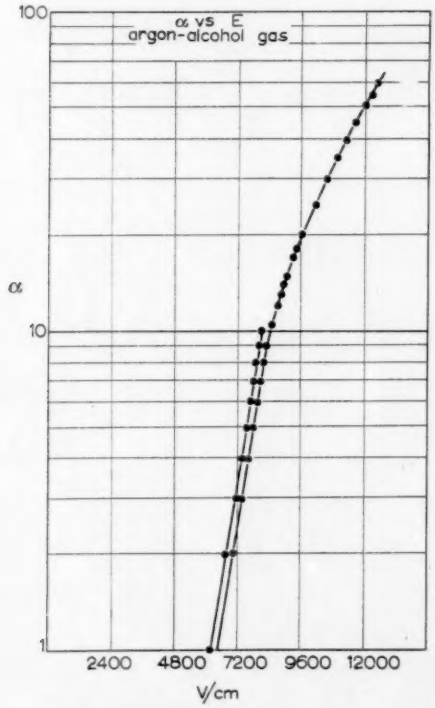


FIG. 1. Gas multiplication as a function of field strength for the argon–alcohol mixture. Upper curve d (separation) = 0.311 mm. Lower curve d = 1.59 mm.

$$J_d = \frac{J_0(e^{\alpha d} - 1)}{\alpha d} \, .$$

Since J_d, J_0 and d can be measured, it is possible to determine α as a function of field strength. Fig. 1 shows the result obtained with a gas mixture of argon and alcohol when two different plate separations (d) were employed.

The separations were determined electrically by connecting one electrode to a rotating potentiometer and by determining the induced current. The accuracy of this method was probably no better than ± 5 per cent. Since d enters into the equation as an exponent, experimental errors of this magnitude could easily account for the difference observed. It is to be noted that α as given in Fig. 1 merely represents the value of an *apparent* first Townsend coefficient. It is possible that the actual processes taking place in the chamber are somewhat more complex than the simple picture presented here.

Chamber design

Fig. 2 shows a diagram of the chamber employed in most of this work. For purposes of detailed studies the spacing of this chamber may be varied from a fraction of a millimeter up to about 3 mm. The top electrode is aluminum foil of approximately 7 mg/cm^2, which is cemented to an aluminum ring. For purposes of mechanical stability a thin rib structure of steel was employed to reinforce the foil along two major diameters. The insulation of the upper electrode is sufficient to permit application of several thousand volts. The collecting system near ground potential is made of electrodes of conducting plastic and is divided into an annular guard ring and the cylindrical collecting electrode that is separated by an annular gap of 0.010 in. width. Since gas purity is of importance, most of the experiments were performed with a steady stream of gas flowing through the device. Employed were tank gases such as argon, methane, carbon dioxide, Q-gas (98.7 per cent helium and 1.3 per cent butane) and P-gas (90 per cent argon and 10 per cent methane). In addition, an argon–alcohol mixture was obtained by permitting argon to bubble through an ethanol column. In all cases the effluent gas was permitted to bubble through a shallow layer of water which insured essentially atmospheric pressure in the chamber. Since for purposes of some of the measurements it was desirable to operate the chambers in a sealed-off condition, the

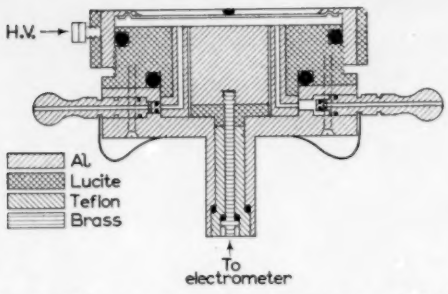


FIG. 2. Diagram of chamber. Gas inlet on left in "closed" position. Gas inlet on right in "flow" position.

fittings shown in Fig. 2 were designed for that purpose. When both fittings are positioned like the one on the left side of Fig. 2, the chamber is sealed; when both fittings are withdrawn by a short distance (with the leaf springs permitted to engage in the inner notches of the fittings) flow may be established. The fitting on the right side of Fig. 2 is in this position. The current was measured by a commercial electrometer (Ultraohmeter). Voltages were supplied from a power supply having at least 0.1 per cent regulation. Since it was necessary frequently to approach and exceed breakdown voltages, and since breakdown tends to damage the electrode surfaces, a special circuit was designed,[3] which lowered the voltage in case of excessive current drain.

RESULTS

Fig. 3 shows typical curves obtained with several gases. It will be shown that the current rises more or less exponentially to a value which is more than 10,000 the saturation value. In some experiments multiplication in excess of 100,000 was obtained before breakdown occurred. Because of the exponential nature of the current versus voltage curve, an equal absolute change in voltage represents an equal change in percentage of the current. The factor depends somewhat on the gas but typically a 10 V increase will increase the current by a factor of about 1.5. Hence, a requirement of 10 per cent stability necessitates voltage stability within about 2 V. The power supply employed was sufficient for stability of somewhat less than ± 5 per cent over the entire range of the curves in Fig. 3.

While it is thus evident that large gains in

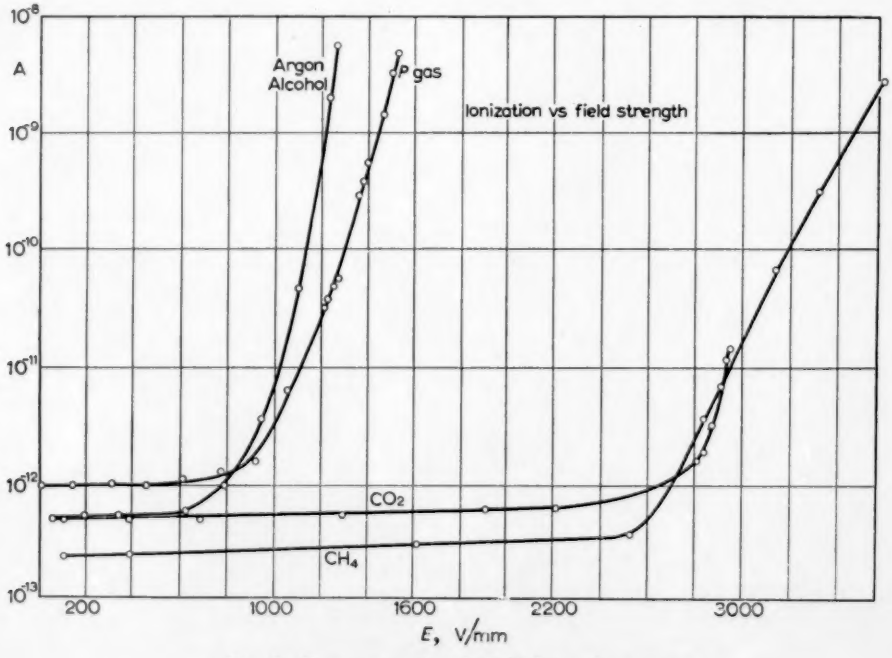


FIG. 3. Current vs. electric field for various gases.

current may be obtained, the device would be of little practical value unless the multiplication is independent of radiation intensity. We have checked the proportionality of this system by employing sources of decreasing intensity under conditions where the decrease was predictable.

One method employed a gamma-ray source which was placed at various distances from the chamber. When the source was near the chamber, currents could be determined with multiplication factors varying from 1 to about 5000. When the source was at large distances from the chamber, the saturation current could not be determined but its value predicted with reasonable accuracy on the basis of the inverse square law. In these instances only the values obtained with at least moderate gas multiplication could be determined. In this way it was possible to cover a thousandfold range of intensity. Because of finite source size and scattering, the I versus d curves did not exactly follow the inverse square law (see Fig. 4) but at all gas multiplications the curvature of the line appeared to be the same.

A second method which was free from any extrapolation errors consisted in placing a beta ray source (Na^{24}) in front of the chamber for the determination of the current versus voltage curve at various times. Since the half-life is accurately known, the saturation current could be predicted even after it was too small for measurement. However, the multiplied currents could be determined readily after the source had decayed for approximately ten half-lives. Fig. 5 shows excellent agreement between observed values and values predicted on the basis of decay.

As a result of both experiments, it would appear that true proportionality is maintained through intensity variations of at least 1000 with multiplications in excess of 10,000.

Experiments were performed with the chamber in a sealed-off condition to determine if impurities would seriously affect chamber operation. As shown in Fig. 6 marked variations did indeed occur. These, at least in part, must be due to air leakage, which is evidenced by the rise of the current values observed without multiplication. We believe that it will be possible to design a considerably tighter system. However, it should be noted that while over a

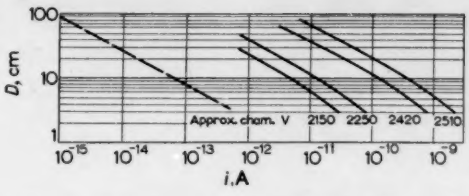


FIG. 4. Current vs. distance from Co^{60} source at various voltages. The broken line represents the slope expected on the basis of the inverse square law. The curvature of the experimental plots (obtained at significant gas multiplications) is attributed to scattering of the gamma radiation.

period of 2 weeks the field required for equal multiplication had to be doubled, the shape of the multiplication curves remained unchanged. Hence, the multiplication attainable is unaltered.

DISCUSSION

Although certain aspects of this instrument may make it valuable for laboratory research, an obvious application is in health physics. If the sensitive volume were of the size of a 25 cent piece, multiplication of 10,000 would yield a current of more than 10^{-12} A for a dose rate of 2 mrad/hr. The requirement of a well-regulated portable voltage supply has already been solved for purposes of survey scintillation counters. If a weak, long half-life radioactive

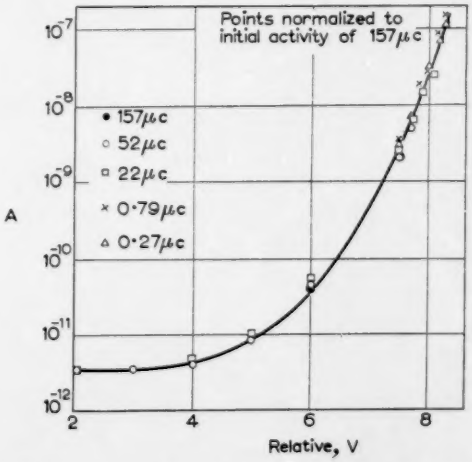


FIG. 5. Current vs. chamber voltage for Na^{24} beta-ray source. Data obtained at various times are corrected for decay.

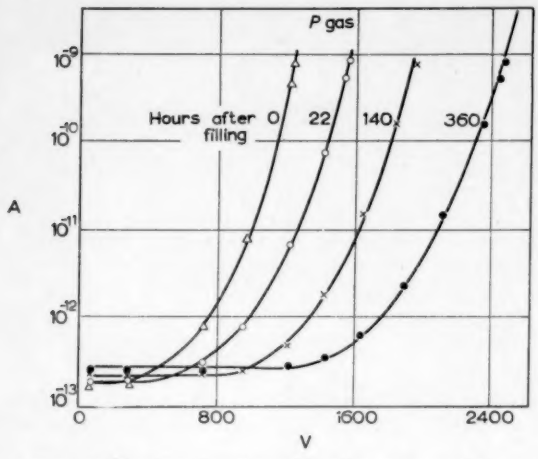


FIG. 6. Current vs. chamber voltage at various times after sealing of chamber. Original mixture in chamber was *P*-gas.

source were employed in a repeatable geometry, the instrument could be calibrated by merely adjusting the voltage until the current meter registers a definite deflection. Most variations of characteristics could thus be eliminated. This calibration procedure would obviate the need for application of a scale factor. While these characteristics would make the device a very useful survey meter for any radiation, it would be particularly valuable in the field of beta-ray dosimetry, where at present no adequate practical instrumentation is available for the measurement of low radiation levels.

Acknowledgments—D. SACHER and M. FELDMAN assisted in many phases of this work and their help is gratefully acknowledged.

REFERENCES

1. G. FAILLA, *Radiology* **29,** 202 (1937).
2. *Annual Report on Research Project* p. 105. Radiological Research Laboratory, Columbia University, NYO-4642 (1956).
3. S. KRAKAUER, *Rev. Sci. Instrum.* **27,** 414 (1956).

Health Physics Pergamon Press 1958. Vol. 1. pp. 51–55

CALCULATION OF RADIATION DOSE

W. S. SNYDER

Health Physics Division, Oak Ridge National Laboratory,
Operated by Union Carbide Nuclear Company for the
U.S. Atomic Energy Commission

(*Received* 30 *October* 1957)

Abstract—This paper considers the influence of phantom geometry on the dose, particularly the maximum dose, within the phantom. It is shown that the maximum dose within an irradiated convex phantom cannot exceed the maximum dose in a slab similarly irradiated and thus maximum doses computed for slabs supply upper estimates of doses for many phantoms. The "first collision dose", i.e. the dose delivered in a volume element by those first interactions of the radiation which occur in the volume element, supplies an under-estimate. For radiations such that the energy released may be considered as locally absorbed, these two estimates are surprisingly close, e.g. for neutrons above 1 MeV and in 30 cm slabs, the difference is about 20–30 per cent.

THE passage of radiation through matter may be described as a kind of "random walk" of a particle through the medium with the possibility of certain interactions of the particle with the medium at each step. If the radiation field is stationary in time the number of interactions of the primary particles with the medium occurring per unit time in a volume element dX and state dV is a function $F(X, V)$ of position and velocity or energy before the interaction. This function is the "interaction density" or "collision density" of the primary radiation. In common with other random walk problems this function satisfies an integral equation,

$$F(X,V)=S(X,V)+\iint dY dU K(Y,U,X,V)F(YU).$$

The function $S(X, V)$ is the source term and gives the density of first interactions of the radiation with the medium, and the kernel K determines the probability that a particle scattering in $dYdU$ will have started in dV after the collision and will scatter next in dX about X. The integration may be considered to extend over all the Y–U space but if backscatter can be neglected the integration with respect to the Y variables may be limited to the actual region where the probability of scattering is not considered to be zero. This particulate mathematical model is convenient in trying to grasp some of the factors which influence the distribution of dose in the irradiated medium. The integral equation formulation enables us to infer quite a few intuitively obvious and useful relations that are often overlooked.

Most obvious is the truism that F cannot be less than S, or that the "first collision" density is a lower bound for the total collision density. Since S is easy to calculate, the energy released by the first collisions is also easy to calculate, and if most of this energy is absorbed near the point of release, a "first collision" dose may be easily calculated. If the irradiated region is sufficiently small this estimate of dose will be a good approximation to the total dose. The first collision dose for neutrons in tissue at a depth T below the irradiated surface of a body is given by equation (1):

$$\text{dose} = \text{first collision dose} + \text{"build-up" dose}$$

where first collision dose =

$$Ie^{-\sigma T}\left\{\sum_i \sigma_i \frac{2M_iE}{(M_i+1)^2}+\sum_i \sigma_i(ab)E_iA_i\right\}. \quad (1)$$

Here M_i is the mass number of an atom of type i, E is the neutron energy, and I is the intensity of the beam (n/cm²). The formula allows for two kinds of interactions—elastic collisions, and

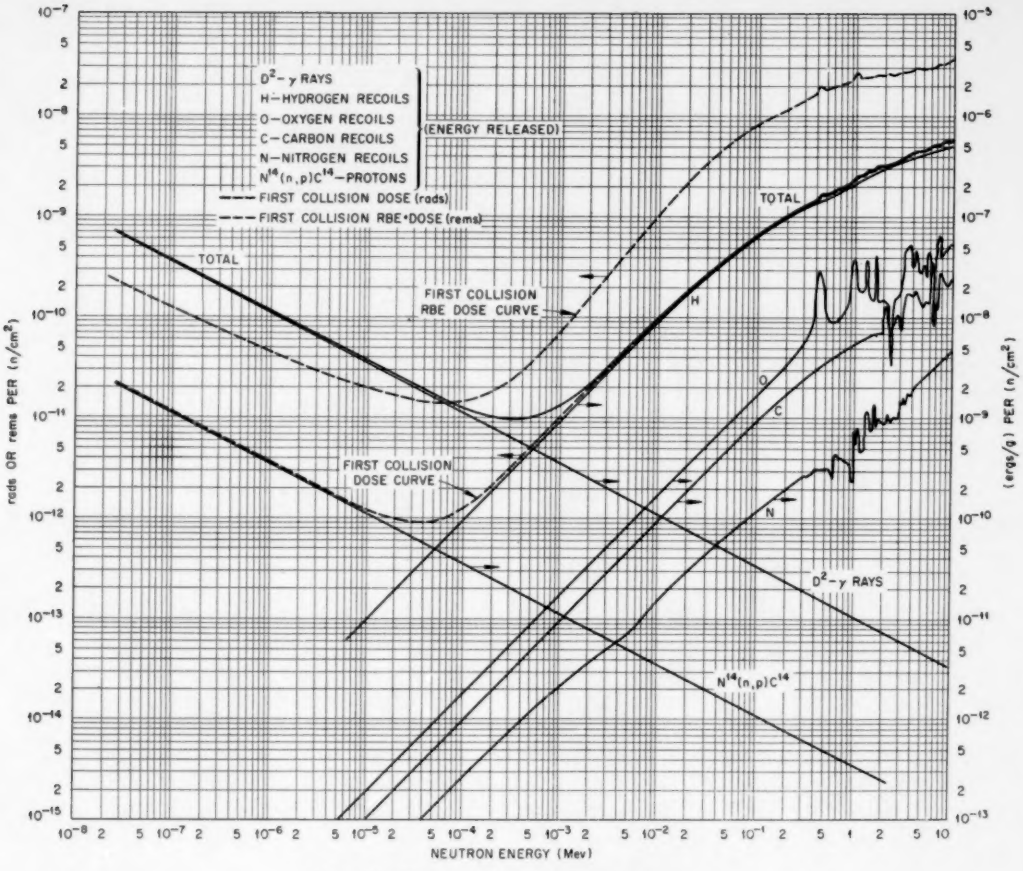


FIG. 1. First collision dose curve.

absorptive interactions with respective macroscopic cross-sections σ_i and $\sigma_i(ab)$ for the ith constituent element of the tissue. In the case of the absorptive interactions the energy E_i released may not be locally absorbed and a factor A_i represents the fraction of this energy that is absorbed locally. The formula is easily generalized for other types of radiation. Fig. 1 shows the variation of this first collision dose with neutron energy in a tissue medium. The energy released by the various neutron interactions in tissue are shown separately as well as the total energy released in the tissue element. The dose represents only the energy absorbed and since the volume element is assumed to be small the energy carried by the photons largely escapes and so is not included in the dose curves. The volume element is assumed to be large compared to the range of the recoils and n–p protons and so these energies are included in the estimate of dose. The dashed curve gives the biological dose, i.e. the dose in rems obtained by weighting the absorbed energy with the RBE values given in Handbook 59 of the National Committee on Radiation Protection.

Fig. 2 illustrates the principle that in general the collision density, and hence the dose, tend to increase with the size of the irradiated object. More precisely, if a convex body is increased by the addition of material which does not shield the original body from the incident radiation, then the collision density and hence the dose are increased. The integral equation shows this very clearly since if the integration is considered

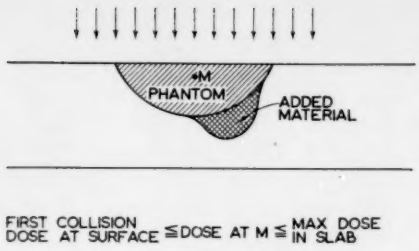


FIG. 2. Variation of maximum dose with the shape of the phantom.

to extend over all Y–U space, then when the region occupied by matter is increased, S and K become positive in regions where they were zero before and are not decreased in the original region. In particular the dose in a convex body cannot exceed the dose in a slab containing the body if the radiation is incident on a nearly plane surface of the body. Thus computations for slabs supply upper estimates of dose for many phantoms. Accordingly, depth dose curves in slabs have been calculated for various media and types of radiation. The National Committee on Radiation Protection's Handbook 63[1] on protection against neutron radiation, contains depth dose curves in tissue slabs for neutron beams with energies varying from thermal energy to 10 MeV. A few of these curves are shown in Figs. 3–6. Fig. 3 shows that for

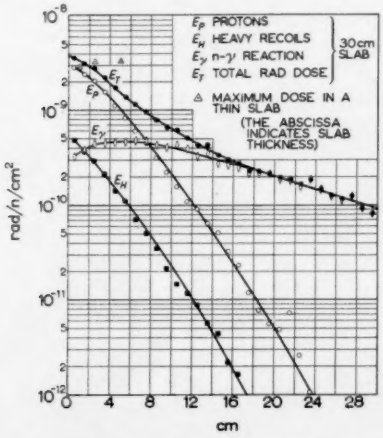


FIG. 3. Absorbed dose (rad) from 1 MeV neutron beam.

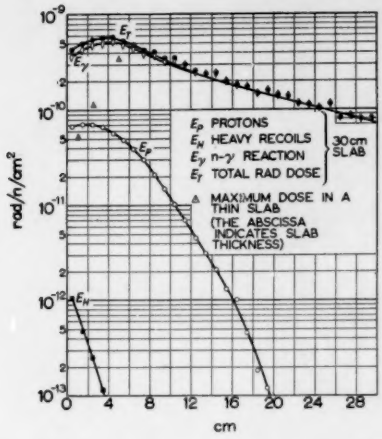


FIG. 4. Absorbed dose (rad) from 0.005 MeV neutron beam.

neutrons of 1 MeV energy, the protons, i.e. recoil hydrogen nuclei together with (n,p) protons, contribute most to the dose near the irradiated surface while deep within the slab the energy absorbed from gamma rays predominates. Note the gamma peak at about 5×10^{-10} rad/n per cm². For neutrons of energy 5 keV (Fig. 4) the energy absorbed from the gamma rays is preponderant throughout the slab. However, the gamma curve is not greatly changed though the peak has moved somewhat closer to the surface. Fig. 5 shows

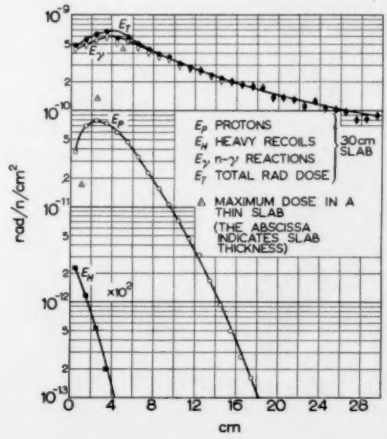


FIG. 5. Absorbed dose (rad) from 0.0001 MeV neutron beam.

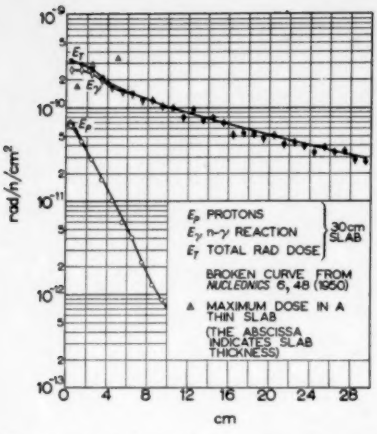


FIG. 6. Absorbed dose (rad) from thermal neutron beam.

for neutrons of 100 eV the same sort of behavior of the gamma component and the sharper peak of the proton component indicates that the n–p reaction is here more important dosewise than is scattering with hydrogen. Finally, Fig. 6 for thermal neutrons, shows the gamma component reduced to about a third of its previous height. Only at these low energies does the penetration become sufficiently small to afford the gamma rays a markedly better chance of escaping. Thus we may say that for thick slabs and large bodies the peak dose due to gamma rays does not vary greatly in size and is, of course, always a broad peak and this peak dose is remarkably insensitive to neutron energy.

Beams have been computed for thinner slabs but the exploration here has been less thorough and much detail needs to be filled in. The small triangles in Figs. 3–6 show peak dose in slabs of 5 cm, 2.5 cm and 1 cm thickness. The triangles are positioned to indicate the thickness of the slab rather than the location of the peak. In general, at high neutron energies, say above 0.1 MeV, the maximum neutron dose does not vary greatly with the thickness of the slab, nor even with the shape of the phantom. This is true since the dose must lie between the first collision dose and the dose in the thick slab and these differ only by 20 to 30 per cent over most of this range. At lower energies the thin slabs give much less dose and this is clearly to be expected since at these lower energies the principal contribution to dose comes from the gamma rays which can easily escape from the thin slab. Even for 5 cm thickness, however, the variation is not excessive. Dosewise, at least, the rat would seem to make a rather good model for a man, particularly if the rats are not irradiated individually but are arranged to form a layer or slab. The results for thin slabs given here are the results of small-sample Monte Carlo calculations; thus they do not have as much precision and detailed information concerning the various components of dose as do the calculations for the thick slabs. They are, however, sufficient to indicate the order of sensitivity of dose to geometry or neutrons of various energies. HOLLAND and RICHARDS[2] have made similar calculations for the scattering of neutrons in air, particularly for a fission spectrum.

Similar calculations have been made for gamma rays in water at the National Bureau of Standards and also at the Oak Ridge National Laboratory. For example, BERGER[3] at the Bureau of Standards has considered a point gamma source at the center of a finite sphere and estimates the ratio of dose at various distances from the source to the corresponding dose in an infinite sphere. He finds the decrease at the surface of the sphere is only of the order of 15 to 20 per cent for the energies tested.

Much remains to be done, particularly for relatively small objects but there is sufficient information presently available to answer some of the questions. Thus a straight-ahead calculation of gamma dose amounts to keeping the cross-section constant and making the photons scatter straight-ahead. If the photon energy is such that the cross-section is larger for lower energies—and this is usually the case—then this clearly gives the energy a greater probability of being carried out of the medium and thus the total dose is underestimated. Monte Carlo calculations with photons of 2.2 MeV energy indicate good agreement with the straight ahead approximation even in thick slabs, i.e. 30 cm slabs. Certainly this is a rather favorable case since the photons do tend to scatter forward at this energy. It would be valuable to set limits on the accuracy of the straight-ahead approximation at lower energies. At present

there are definite limits to our ability to calculate dose. For example, cross-sections for inelastic scattering are not well enough known to permit calculations in many cases. But this situation is probably short term. The difficulty of the calculations seems, in general, to require elaborate facilities for computation. Perhaps we should not expect to get precise answers, computed to order, for every problem we encounter. But it is quite feasible to write a fairly general code to compute a certain class of problems, and as high speed computers become generally available we may expect such a development. In the interim we can still obtain much information from typical cases that have been computed, and the general principles governing the interaction of radiation with matter can do much to adapt this knowledge to the specific needs of the experimentalist or the health physicist.

REFERENCES

1. *Protection Against Neutron Radiation up to* 30 *Million Electron Volts*. National Bureau of Standards Handbook 63, Superintendent of Documents, Washington, D.C. (1957).
2. S. S. Holland, Jr. and P. I. Richards, *J. Appl. Phys.* **27,** 1042 (1956).
3. M. J. Berger and J. Doggett, *J. Res. Nat. Bur. Stand.* **56,** No. 2, 89 (1956) (Research Paper 2653).

Health Physics Pergamon Press 1958. Vol. 1. pp. 56–61

WHAT IS HEALTH PHYSICS?

WALTER D. CLAUS
Division of Biology and Medicine, U.S. Atomic Energy Commission,
Washington 25, D.C.

(*Received* 4 *December* 1957)

Abstract—The science of health physics is charged with the task of providing protection to all living things against the potential hazards of radiation, while at the same time making it possible for the human race to enjoy all the benefits which may arise from the use of atomic energy. "Health physics" is a term which came into being in the Plutonium Project about 1942, and which has now come to be used almost universally to mean the science or profession of radiation control in all its many aspects, including technology, philosophy, economics, labor relations, public relations, teaching and administration. Scientifically, it has common interests with a number of well recognized fields of specialization: physics, electronics, biophysics, chemistry, biochemistry, biology, physiology, genetics, toxicology and ecology.

Some of the areas in which the present-day health physicist finds himself participating are:

(1) The development of sound philosophy and principles of radiation protection.

(2) Application of these principles to practical situations (e.g. participation in the design of hazardous experiments, monitoring of personnel and of sites, shielding, waste disposal, shipping, etc.).

(3) Devising methods and techniques of dosimetry, with suitable instrumentation for implementing them, based on knowledge of physical and biological facts about radiation.

(4) Giving assistance, advice and cooperation to constituted health authorities, in the protection of community health against indiscriminate handling of radioactivity.

(5) Assisting in the education of workers and of the general public in fostering a better understanding of the hazards (and lack thereof) of radiation.

(6) Assisting in civil defense and emergency planning.

Training courses in health physics are available in a number of our universities, but health physicists will find that they need to draw on all the knowledge and experience at their personal command, will need to weld it with what they learn in course work, and then face each radiation protection problem with judgment and ingenuity. Health physics is a fascinating and challenging field. Good men are sincerely to be encouraged to enter this promising profession.

THE term "health physics", for want of a better expression, has come to be used almost universally to mean radiation protection, radiation hygiene, radiation safety or radiation control. Physics, biology and chemistry provide the technical foundation for the science of health physics, but many ancillary activities are associated with it—philosophy, economics, labor relations, public relations, teaching and administration.

Health physics includes within its protectorate all the world of living things. It shields the people who work with radiation; it guards their families and friends, and the general populace who may suffer from the indiscriminate use of atomic energy even as they stand to benefit by its development; it protects their children, their grandchildren and all their progeny to come, from mutation-inducing radiation; it protects their livestock, their crops, their soil and its productivity from undue radioactive contamination; it guards the ecology of all things that live on the land and in the sea from this new disturbing influence.

Its dominion is our total environment. It extends from the ooze beneath the sea, through the hydrosphere, the lithosphere and the troposphere, and reaches far into the stratosphere.

In short, it is just about all things to health physicists whose job it is to provide protection against the potential hazards of radiation, while at the same time making it possible for the human race to enjoy all the benefits which may arise from the use of atomic energy.

History of health physics

Historically,* the term "health physics" had its origin in the early days of the Plutonium Project, in the United States. In 1942, when the scientists became reasonably certain that they could produce a nuclear chain reaction, they also became aware of the biological hazards of their accomplishment. They realized that the fissioning of even small amounts of uranium would produce the radiation-equivalent of more radium than had ever been concentrated on earth. The very process of fissioning would produce terrific fluxes of gamma rays and of even more hazardous neutrons. And worse, the resulting fission products would be a legacy requiring care almost forever.

Thus there was called into being in the Plutonium Project a health division, with a medical section, a biological section and a physics section. It was quite natural to call this last section "health physics" to distinguish it from the physics of other divisions, such as chemical physics. The term "health physics" was used for those activities in which physical methods were used to determine the degree of radiation hazard to the health of the project personnel. It was necessary to measure the amount of radiation, to characterize it, and to interpret the physical readings in terms of the biological damage which radiations do to living things. Physicians and biologists might determine the actual health and biological changes, but physicists had to determine what kind and how much radiation it took to produce those changes.

One of the first problems of health physics was to calculate how much shielding would be necessary about a reactor if the estimated radiation intensities were anywhere near correct. It was not the simple matter of applying known absorption factors for gamma rays in concrete. Broad beams of radiation are absorbed differently than narrow beams, and absorption coefficients in many shielding substances were poorly known for both gamma rays and neutrons. Nor was it known that little-understood particles such as neutrons and mesons might not also be present in large quantities.

Chemical separation cells also needed to be shielded, as well as the laboratories for analyzing the materials. Complete shielding was impractical, for materials had to move in and out, with possible hazard to the workers. So methods for the monitoring of personnel had to be devised and evaluated. Badges containing photographic film, small pocket ionization chambers, and portable radiation-measuring instruments of all kinds were evolved. To the great credit of these early health physicists, they produced useful instruments in a few short months, in a field which still commands the attention of dozens of skilled instrument designers.

Work areas had to be monitored. Spills were bound to occur, and radioactive contamination could spread throughout work areas and laboratories, and over the workers. Monitoring methods, decontamination procedures for use on men and on their clothing and equipment, special laundries, all had to be devised.

The public at large also had to be protected from the damage which might occur as a result of the operations of the plant. Thus the atmosphere for miles around had to be monitored, and the background radioactivity in the communities checked against undue increase. Gases and vapors passing from the chimneys into the atmosphere, and waste waters going into the rivers, were monitored with specially devised, high-sensitivity instruments. At that time there were no really satisfactory electronic instruments available commercially, so that the health physicists were faced with the problem of developing and making practically all the instruments that they used.

The health physics section had also the tremendous problem of organizing itself in such a way that proper records would be retained on

* The writer is indebted for much of the following historical discussion to the paper of Dr. R. S. Stone, *Proc. Amer. Phil. Soc.* **90** (1946).

the radiation exposures of all individuals, and on the radiation levels in all parts of the facilities and in the air and water. Moreover, as the project grew, so also grew the need for more health physicists and technicians—teaching and training became a sizeable task.

The health physicist was the person best acquainted with the behavior of the radiations and of the persons handling them, so that he was in an advantageous position to give advice on laboratory and production processes. He thus came to participate in atomic energy operations in a positive way as well as in a negative way—that is, to help set up processes that were safe, rather than simply to protest that they were unsafe.

Health physics today

Now we have seen how health physics came into being, and how it got its name. It started out as physics applied to the control of radiation, but health physicists soon demonstrated their usefulness in areas other than physics alone.

The field of nuclear science and atomic energy has developed rapidly since man initiated that first chain reaction in Chicago on 2 December 1942. Along with this development has come a vast increase in the amount of radioactivity to which man and his environment may potentially be exposed. Ionizing radiation in sufficient quantity is known to possess not only beneficial properties, but also the power to do great damage to living organisms. Reactors, high-energy accelerators and the use of radioisotopes in research and medicine—all present potential hazards not only to the workers themselves but also to the populace as a whole should the environment become unduly contaminated. The number of these problems may be expected to mount with terrific impact as more and more power-producing reactors come into being.

Thus there has arisen a need for persons who understand these hazards and how to prevent them, yet who have sufficient knowledge of the atomic field that they can work with other scientists and engineers in promoting useful developments. That is, their approach must be positive, to promote the benefits of atomic energy, as well as negative, to prevent damage to man and his environment.

Health physics has common interests with a number of well recognized fields of specialization: physics, electronics, biophysics, chemistry, biochemistry, biology, physiology, genetics, toxicology and ecology. It may be defined as the *science or profession of radiation protection* in all its aspects.

The student of health physics must obtain fundamental knowledge in radiation physics and biophysics, interaction of radiation with matter, interpretation of physical measurements, instrumentation, an understanding of the principles of permissible exposure and the prevention of undesirable exposure, and an introduction to the legal and public relations aspects of radiation protection.

The health physicist will in many cases be closely associated with research programs, and may often have the opportunity to participate in research work, or in laboratory and committee work leading to improved permissible levels of radiation exposure, and to better methods and instruments for assuring radiation safety. By reason of his opportunities to participate in production activities—production of materials or power—and his close association with personnel, he often attains a managerial capacity.

Let us consider briefly some of these areas in which the present-day health physicist finds himself participating.

Development of sound philosophy and principles of radiation protection

If the primary job of the health physicist is to protect man and his environment from the detrimental effects of radiation and of radioactive substances, his very first task is to determine the degree of protection which must be provided. Man cannot work with radioactivity without exposing himself to some extent, nor without releasing some of the radioactivity to his surroundings. Is there a certain amount of such exposure which he can tolerate, without expectation of bodily harm—or, perhaps more properly expressed, without expectation of more harm than he can afford to suffer in return for the benefits to be expected? Certainly some radiation is tolerable, for mankind has developed through hundreds of thousands of years of exposure to low-level background radiation.

Certainly large amounts of radiation are intolerable, for it is known that man can be killed by a few hundred roentgens of X-rays or gamma rays administered in a short time. Somewhere between these two levels is a dose rate which, even if absorbed over many years of working life, is a tolerable hazard to accept in return for a comfortable living. Note the term "tolerable hazard" for it is believed by many that *all* radiation exposure carries with it some degree of hazard. Thus the health physicist, in association with scientists in many disciplines, must agree upon a "maximum permissible dose" for radiation workers and for general populations, which he believes is quite safe as compared with the other hazards of living. From this figure for maximum permissible dose, and from other biological and physical data, the health physicist can calculate the limits which must be put upon the permissible degree of contamination of work areas, of skin and clothing, or air, water and food.

Now armed with these basic criteria of permissible exposures, the health physicist is in a position to ascertain how to enforce these limitations. Data, codes, guides and suggestions are available in the handbooks and reports of authoritative groups. The health physicist must know this material and how to apply it intelligently in the conduct of his job of radiation control.

Application of principles to practical problems

The application of the principles to practical situations through suitable methods and techniques (e.g. participation in design of hazardous experiments, monitoring of personnel and of sites, shielding, waste disposal, shipping problems, etc.) can be thought of as the true backbone of radiation protection. Whether workers are conducting experiments or doing routine production work about a reactor or in a chemical processing plant, their actions must be controlled in relation to the radiation sources with which they are dealing, in relation to the nature and intensity of the radioactivity, and to the shielding which can be provided. The degree of success of the health physicists' protective efforts inside the plant or laboratory is determined through monitoring of the workers and of the work areas. But people living outside the plant or laboratory cannot be conveniently monitored; hence the degree of their protection is determined by monitoring their air and water, and perhaps also their food and soil. Furthermore, good records must be maintained on all these monitoring observations.

All atomic energy plants and laboratories produce wastes of some kind. Some are gaseous and can be allowed to escape up tall chimneys where they are diluted with large volumes of air and so rendered innocuous; or they may be filtered and scrubbed to remove hazardous particles to such an extent that the remainder may be safely discharged. Water-borne laboratory wastes are usually of relatively low concentration in relatively large volumes of water. Sometimes this waste can be disposed of directly to the soil, or to rivers and streams without hazard to the subsequent users of those waters. Sometimes the waste must be treated to remove radioactivity before discharge, or simply stored long enough to permit radioactive decay to reduce the activity to a safe level for discharge.

But the disposal of high-level wastes from chemical processing plants constitutes a problem that is not yet solved. To date such wastes have only been stored in tanks—an impossibly expensive procedure when we think of process wastes still to come from power reactors. Methods are being sought for binding these wastes in an insoluble manner with sintered minerals, so that they can then be safely buried in the soil or in the sea. Studies are being made to determine the feasibility, from the health and safety viewpoint, of disposing of these high-level wastes directly into certain geological formations and into deep-sea areas. Without doubt, the whole question of waste disposal is the knottiest problem which the health physicist has to handle.

The shipping of radioactive materials is another problem of importance. Ores can be handled easily but more concentrated extracts present problems of cleanliness. Spent fuels containing fission products, and/or these same materials after chemical processing, constitute a great hazard to the countryside when they are shipped by truck or freight train from plant to plant or to disposal areas. Simple maintenance of the vehicles is hazardous enough, but wrecks are unavoidable from time to time. Spills of

large quantities of radioactivity on roads or railways, or into water-sheds, constitute the shipper's nightmare.

Development of techniques and instruments for dosimetry

We have already seen how the early health physicist found it necessary to devise and build the instruments and devices he needed, in order to measure and characterize the radiations with which he had to cope. Geiger counters, proportional counters, scintillation counters, ionization chambers, tissue-equivalent chambers, quartz-fiber electrostatic chambers of unbelievable sensitivity and ruggedness, magnificient photo-multiplier and electronic tubes—all these and many more have been the result of the health physicist's need, and often of his own ingenuity.

The health physicist has needed to know how radiation of all kinds delivers its energy to the tissues and organs of the body, and he has needed to know how to measure these delivered doses. Gamma rays, alpha rays, beta rays, neutrons—all affect living things in different ways which the health physicist must know about in order to build his instruments and determine the doses.

But even today, after fifteen years of intensive study in the fields of instrumentation and dosimetry, there remains much still to be accomplished. Radiation in practice is seldom mono-energetic. It usually comes in a mixture of kinds and energies. The problem is to unravel its complexity and to make meaningful measurements on a simplified basis. Great advances have been made. Many more remain to be made, and the health physicist with a bent for instrumentation will find plenty to occupy his interest.

Bio-assay is a dosimetry technique which may be mentioned just in passing, with a brevity quite out of proportion to its importance. Radioisotopes may contaminate air, water, and food, and they affect the living organism as they pass through it. In addition, the human or animal body selects out certain of these radioisotopes for retention in one organ or another. Thus, radium and strontium are retained in the skeleton because their chemistry is similar to that of calcium. Relatively insoluble particulates may be retained in the lungs for considerable periods of time. Others will be retained in the liver, or in the lymph nodes.

Much effort is going into the problem of determining how much radioisotope is permissible in the body, but an additional problem is to determine how much is actually in the body, as for example after an accident or after prolonged exposure to low concentration in air or water. This determination is known as bio-assay. It may be accomplished in some cases by physical methods, such as by measuring in a special "whole-body counter", or with a strategically placed detector, the radiation emitted by the isotope in the body. But more often, bio-assay is accomplished by radiochemical analysis of excreta, and calculation of body content from previously determined rates of elimination of the isotope by the body. Bio-assay is one of the lesser developed, but nevertheless one of the most important, personnel monitoring methods available to the health physicist.

Co-operation with public health agencies

The health physicist, by reason of his highly specialized training, has responsibilities in his community beyond the simple performance of his technical duties within the confines of his plant or laboratory. Health authorities in a community—be it city or state—are concerned with the effects of radioactive effluents and contamination upon the health of the community just as they are concerned with the effects of many other kinds of pollutants. In the present state of knowledge—and lack of knowledge—about radioactivity, many people have become unduly fearful of radiation. Most health authorities have not had an opportunity to study adequately the true nature of this new hazard. Only a few health departments have had the funds and the good fortune to add a competent radiation expert to their staffs.

As a consequence, one of the most far-reaching services that a health physicist can render to his community is to maintain a close liaison with the public health official and to discuss in the fullest possible manner the

accepted concepts of permissible dose, his problems and his methods of radiation protection.

Educational activities

Great alarm has been raised in people's minds during the past few years by incompetent discussion of the hazards of radioactivity. Oddly enough, good-housekeeping practices also can cause alarm among radiation workers. The health physicist normally tries to keep work areas at the lowest practicable degree of contamination, primarily in order to be able to detect quickly any leaks or aberrations in the normal work processes. Levels of activity many times higher may have no significant health hazard connected with them, yet the emphasis on maintenance of low levels often gives workers a morbid fear of somewhat higher levels. The same situation may prevail with people in the environment of a reactor or a laboratory.

The competent health physicist will develop sound concepts and philosophy in this matter, and will arrange means of discussing the questions with workers and with the people living in the area. He will hold discussions and conferences with his workers, and give lectures on suitable occasions to schools, clubs and other groups in his community.

Civil defense and emergency planning

Emergency planning is another community service in which the health physicist can assist. Civil defense throughout the world is concerned in a greater or lesser degree with considerations of nuclear warfare. The health physicist has high competence in areas of radiation effects, shelter, measurement and decontamination. These same considerations come into the picture when one thinks of the situation which might prevail in the unlikely event of a large reactor disaster. The health physicist may expect to find his special competence in high demand in such emergency planning.

CONCLUSION

If now one reviews all these varigated activities associated with the profession of health physics, it is obvious that health physicists cannot be produced by a few months, or even by a few years, of instruction. Courses will give to students an introduction to some of the tools that are needed—the technical information of the profession. But health physicists will find that they need to draw on all the knowledge and experience at their personal command, will need to weld it with what they learn in course work, and then face each radiation protection problem with judgment and ingenuity. Whether they had once thought to be physicist, chemist, medical man or public health officer, biophysicist or engineer, they will find specialties in health physics to stir the interest and imagination. Good men are sincerely to be encouraged to enter this promising profession.

Health Physics Pergamon Press 1958. Vol. 1. pp. 62–70

RADIATION EXPOSURE AS A REASONABLE CALCULATED RISK*

LAURISTON S. TAYLOR
Chief, Atomic and Radiation Physics Division, National Bureau of Standards, Washington 25, D.C.

(*Received* 4 *November* 1957)

Abstract—Maximum permissible exposures for radiation workers have been based, in the main, on levels that would not produce detectable injury to any individual. The impossibility of correlating radiation effects with small radiation dosage makes it impracticable to use this criteria indefinitely. It must be assumed that there is some undesirable effect from radiation exposures, even though these are not specifically detectable. We are thus confronted with the necessity for accepting some risk due to radiation exposure, even though this risk cannot be specifically defined. The addition of a risk philosophy in the establishment of radiation protection standards and the acceptance of occupational exposure becomes a necessity, if we are to continue the proper exploitation of radiation as a servant to man. Some comparison of the risk due to various types of radiation exposure is given.

THE lecture this evening will be based mainly on what are believed to be the views of the International Commission on Radiological Protection, the International Commission on Radiological Units and Measurements, and the U.S. National Committee on Radiation Protection and Measurement. The main purpose of the discussions will be to explain the background of the problem of protection against the harmful effects of radiation. With an understanding of this it should be easier to apply the protection measures prescribed by various responsible bodies for the solution of specific problems. Protection cannot be adequately supplied by rule alone—any rules should be regarded mainly as guides, and should be freely supplemented by good judgment, common sense and a full understanding of the end objectives.

Most radiation workers are aware, at least in a general way, of the existence of so-called "maximum permissible limits of radiation dose"; what is not so certain, is their understanding that these levels are to be regarded as maximums, and not as levels which should be applied without further question to the whole population or even the much smaller group of radiation workers.

For the last 30 years the International Commission and related national bodies have, without exception, emphasized the need for working as far below the maximum as circumstances permit. Part of the reasoning behind this emphasis was the strong feeling, based upon very limited evidence, that it might be necessary to establish lower permissible levels when the fraction of the population exposed becomes substantially larger. That time is now approaching. Another part of the reasoning was based upon sheer prudence, with the knowledge that aside from medical applications, no radiation is good for man.

The possibility of the lowering of the permissible exposure levels has been mentioned. Such lowerings have occurred in several major steps since the first quantitative limits were established by the ICRP in 1934. The reasons for this have been various and, except for the most recent change, will not be discussed in any detail. Suffice it to say that the principal factors have been, first, an improved knowledge of radiation effects on the human system and second, the engineering developments that have made it

* Evening lecture presented at UNESCO International Conference on Radioisotopes in Scientific Research, Paris, 10 September 1957.

possible much better to confine the undesired radiation, in spite of the much greater quantities being used.

It may perhaps sound startling to contend that the establishment of permissible levels of radiation exposure is not basically a scientific problem. Indeed, it is more a matter of philosophy, of morality, and of sheer wisdom. True, the basic knowledge used in establishing radiation levels is derived from scientific sources, and by and large the use and application of this knowledge must be by scientists, drawn from many fields of endeavor. Also, the achievement of the protection goals is purely a matter of science and engineering.

But before a maximum permissible dose of radiation can be set for man, there must be an unequivocal answer to one question: "what amount of radiation may man receive in either chronic or acute exposure without any harm to himself or his progeny?" At present the only answer to this question is: "none". There is today little or no direct, positive proof that there does or does not, exist some level of exposure below which harm will not result. It was recognition of this lack of proof that led the ICRP many years ago to discard the term "tolerance dose" which implied that there was some degree of irradiation that was wholly without harm. In its place is used the term "permissible dose", which, while not completely unobjectionable, does not carry the connotation of absolute safety.

There are then, two factors that remove radiation protection from the field of science to the field of philosophy: (1) lack of a clear and scientifically unobjectionable answer to the question posed above, and (2) the consequent necessity for assuming that any amount of radiation, no matter how small, is harmful.

In spite of the lack of direct knowledge of radiation effects on himself, man has recognized and become accustomed to the advantages that radiation offers him, both as to his health and his material well-being. These will not be given up lightly—probably will not be given up at all.

The problem centers about a philosophy to which we are all well accustomed; it may be called a "philosophy of risk". In nearly every phase of ordinary life we accept, knowingly if we think about it at all, risks of commission that could cost us our lives. These risks are accepted in the belief that the possibility of gain offsets the chance of loss. One naturally makes every attempt to reduce the element of risk to a minimum, but in rare instances only can it be reduced to zero. To attain zero risk it is almost certain that life itself would have to be given up. We are thus continually in a situation of having to effect a suitable compromise between risk and gain. The effectiveness with which we can weigh conditions and arrive at a compromise, depends upon our knowledge of both items—risk on the one hand, and gain on the other.

For emphasis we can mention some of the risks that we commonly accept; virtually any mode of transportation from boat or bicycle to the airplane—but without these, life today would be very different indeed; smoking, a luxury that we could well do without—except that we would probably find some other relaxing habit that would be just as dangerous; many forms of medication to which there are dangerous individual idiosyncrasies or allergies.

In essence, radiation risk is no different from the rest of these except for the very important aspect of its probable mutagenic effect that will continue for generations. But there are accepted risks of other nature which may also have some small mutagenic aspect.

With all our lack of direct knowledge of the biological effects of radiation, it is still fairly clear that radiation offers direct advantages to our health or material well-being. In the area of health, it is certain that radiation techniques have contributed most importantly to our greatly increased life expectancy. But simultaneously we also have antibiotics, drugs, surgery and so on to the point where it is probably impossible to ascribe cause and effect to any one. In industry, radiation as an examining tool, is used to detect the imperfections of parts, the failure of which might cost lives. In research, radiation has offered whole new avenues of approach to the solution of problems essential to our growth of knowledge. The advent of nuclear power may have come just in time to alleviate critical power shortage in some countries, and create sources of power in countries that had no power at all. But each of these uses has its own quota of risk

to someone. Are they therefore to be given up? I think not, but there should be a continuing struggle to push the risk to still lower levels.

In comparison with the gains, the risks from radiation use are not very clear and are particularly difficult to define on a cause and effect basis. Except in relatively rare instances, there is almost no knowledge relating death, shortening of life expectancy or genetic abnormalities in man to a specific dose or a particular radiation exposure. It is unfortunate, for analytical purposes, that practically any radiation effect can also be caused by other means, known or unknown. Most radiation effects can be detected only by statistical studies on relatively large population samples. To many persons, grown accustomed to risky living, statistical deaths have little interest until they themselves become one of the death statistics, and then of course it is too late. Who worries because he carries in his pocket two personal death warrants—a pack of cigarettes and an automobile driver's licence? This is mentioned, not to soften any fears but to emphasize the complexity of the problem and its relationship to good judgment and wisdom.

Let us consider very briefly the two main general types of radiation damage to man—somatic and genetic. Until about 10 years ago, the main emphasis was given to somatic damage —that is damage to the body cells of the individual exposed. The basic protection criterion had been to set the exposure levels sufficiently low so that no detectable damage to the individual would result. The possibility of genetic damage was not overlooked, but too little was known about it to permit of its sensible evaluation. Particularly in the light of present knowledge there is a very real question as to what constitutes "detectable damage". Certainly a skin lesion is detectable but a similar lesion within the body might pass undetected forever. If undetected damage exists and the individual dies from seemingly normal causes, who is to say whether or not that damage may have contributed in some obscure way to that death? In fact there is increasing evidence based on animal experiments that there is such a relationship.

Up to now there has been little, if any, acceptable evidence that chronic exposure to radiation at the levels prescribed since 1934 has resulted in definable or detectable damage. It would appear, but no more than appear, that the permissible exposure levels have always been adequately low from the point of view of specific somatic damage.

The possibility of genetic damage has been known to us for some 25 years, but, as already mentioned, our knowledge was very qualitative until a few years ago. Even now most of it is based on animal experiments—human experience is extremely limited—but the burst of effort in studying the genetic effects of radiation over the past 10–15 years has substantially enhanced our knowledge of the subject. Particularly within the past few years the emphasis on genetic effects has increased enormously—some fear it to be out of proportion to the importance of other radiation effects.

The treatment of the genetic problem is one requiring the utmost in wisdom and morality. For, whatever the effect of radiation on our genetic status may be, its principal impact will not be upon the generation of individuals exposed. Rather will it be distributed over future generations up to perhaps 50 in number. Whether the effects on these future generations are to be good or bad—and on the average the prediction is bad—the control of that future lies in our hands. The dilemma is complicated by the fact that there may never be a means by which the cause and effect can be clearly established. Deleterious mutations due to irradiation of persons at presently accepted levels may only be a fifth of those resulting from all causes—the other four-fifths may be spontaneous, or of unknown cause. To what degree are we to be held responsible for our known contribution to this legacy for the future? What is our moral obligation in this regard? The answer is clear: To be morally right, ignorant though we are, we must do all in our power to reduce genetic damage to a minimum, commensurate with our continued use of radiation as an immeasurably valuable servant to man. A workable balance between risk and reward must continually be sought, and is being sought. The recommendations of the ICRP in 1956, based on their earlier findings beginning in

1952, reflect this balance within the scope of present knowledge and technical capabilities.

The geneticist contends that zero radiation exposure to man, apart from natural background radiation, would be the ideal goal, and though as yet unproven he may well be right. This of course means the complete discontinuance of all radiation use—an unacceptable alternative. The problem thus confronting us is to determine or, perhaps better, to decide how much above zero we can operate. The levels presently recommended by the ICRP reflect the best decision on this delicate question that can be reached today. Tomorrow it may be possible to make a wiser decision.

The newly proposed levels of radiation exposure represent a compromise between what we want to accomplish and what we can accomplish while still retaining the necessary uses of radiation. They will introduce temporary hardship to operators in some areas, but on the whole will not seriously interfere with major and worthwhile uses of radiation. They will involve some re-education in the technological world, and some new education for the public. The fact that we can lower our permissible exposure levels by a factor of three below those of 10 years ago is a tribute to the world's radiation industry. Its technological advances over that period have been one of the most important factors in lowering these levels.

Let us consider for a few moments the permissible exposure levels over the past three decades (see Table 1). The latest levels recommended by the ICRP in 1956 average about 5 rems/year for occupational exposure.* From 1950–1956 the average was about 15 rems/year; from 1934–1950 the average was 60 rems/year (in the U.S. the National Committee on Radiation Protection in 1935 recommended 30 rems/year); prior to 1934 the levels were of the order of 100 rems/year. Yet at none of these levels has there ever been developed any positive evidence of damage to the individual. In the main, these successive lowerings represent improved compromises between goal-zero, and capability. Again capability has been made possible by technological advances.

Table 1. *Permissible occupational exposure levels*

	Level (rems/year)
ICRP (April 1956)	5
ICRP (1950–1956) at 0.3 rem/week	15
NCRP (1935–1948) at 0.1 rem/day	30
ICRP (1934–1950) at 0.2 rem/day	60
Prior to 1934	100

Of course, the objectivity of our damage criteria is marred by the possibility of a shortened average life span which, if real, must be classed as damage. Recent studies of the life shortening of radiologists are now considered by some to be inconclusive in the quantitative sense, but it is generally believed that there must be some such effect on man. The estimates vary so enormously as to cast doubt on all of them.

In all of the considerations by the ICRP there has been no assumption of any kind of threshold level below which specific or general radiation damage does not occur; in the absence of any positive knowledge, this seemed the only acceptable conclusion. This is not to imply that there may not be some specific thresholds, though at the moment the possibility seems remote. Statistical data such as that recently obtained for leukemia incidence, while not completely free from objection, points to the conclusion that there is no threshold.

Thus all indications seem to point to the fact that present and future solutions of the radiation protection problem will have to be based on a risk philosophy, will have to be compromise solutions, and cannot be solved on the basis of scientific evidence alone.

At this point one may well ask why so much emphasis has been placed on a discussion of the background philosophy of radiation protection. The purpose has been to impress upon the reader, the fact that protection against the harmful effects of radiation is not a straightforward problem for which there is a simple black or white solution. A degree of protection can be

* The rem is a unit of biological radiation dose and applies to all kinds of radiation including neutrons. It is the product of a measured absorbed dose and a factor called the "relative biological effectiveness". For X-rays and gamma rays 1 rem is essentially equal to 1 roentgen.

had—the exact degree depends upon each of us who uses radiation. Rules provide some basic guidance as to limits which may be regarded as acceptable today, but should never be considered as ideal. Continuous efforts must be applied towards working with lower levels—as low as possible commensurate with the task. The extent to which these efforts are successful may very well affect the whole future use of radioactive isotopes and nuclear power.

Let us now examine the new permissible radiation exposure levels as recommended by the ICRP in April 1956. These new recommendations have not been made capriciously, but have resulted from studies begun in 1952, at which time the ICRP had arrived at essentially the same conclusions as those reached 4 years later. However in 1952 they felt that existing data were inadequate and the problem insufficiently urgent to warrant hasty action. While our knowledge has not increased enormously since then, the urgency has increased to some extent because of the more widespread use of radioactive materials, and because of the public concern over fall-out and the development of nuclear power.

The new levels have been dictated mainly for reasons of genetic damage, but it must now be recognized that somatic damage possibilities are assuming equivalent importance. As already mentioned, the direct evidence of genetic damage to man is almost totally lacking, and that of somatic damage is very scant at best.

In terms deliberately general, the ICRP has made two principal recommendations on the maximum permissible dose: (1) for the population as a whole, the average per capita dose should not exceed 10 rems, from conception up to age 30 (and by inference one-third of this amount per decade thereafter); (2) for radiation workers, the individual occupational dose should not exceed 50 rems up to age 30 (and by inference again this same amount per decade thereafter). It is obvious that these need further interpretation. For that purpose the most recent interpretation made by the National Committee on Radiation Protection in the United States, will be discussed. Interpretations by other countries may differ.

Dealing only with the problem of occupational exposure, an allowance of 50 rems up to age 30 implies an average of about 5 rems per year, assuming 10 working years. For fairly obvious reasons it is impractical to integrate a person's dose over a period as long as 10 years. Since genetic damage was a primary consideration in setting the new levels, it is important to limit the gonadal dose as much as possible in the years during which most children are conceived. Thus it would be desirable to delay the exposure as long as possible—at worst to distribute it nearly uniformly at the rate of 5 rems per year.

This solution however, is not without its difficulties because our legal machinery is accustomed to dealing with the individual and not with averages. The difficulty in dealing with an "average number" is introduced because of the fact that the safety recommendations of a few years ago are now becoming legal requirements backed up by varying degrees of enforcement or pressures. During the 25 years beginning in about 1930, radiation safety the world over, was based on the ICRP recommendations backed up by voluntary compliances and good sense. It must be admitted that there have been times when neither compliance nor good sense prevailed, but in the vast majority of situations they did. The tendency now is for governments to call for strict codification and the development of radiation protection regulations wherein a situation is either black or white. Here begins our trouble.

If we say an average dose of 5 rems per year, but not more than 50 rems in, say, 10 years, we mean just that. We know there is little difference if a person is allowed to receive, say, 7 rems in one year and 3 rems in the next as compared with 5 rems in each year. On the other hand it would clearly be wrong to allow a person of age 28 to receive, say, 25 rems in each of 2 successive years. It would be even more wrong to deliver such doses to a person, say, of age 20.

To illustrate the difficulty, suppose we just say an average of 5 rems per year. Legally this will be interpreted as no more than 5 rems per year. Everything must be black or white and any individual who has been so unfortunate as to have recorded against him a dose of 5.001 rems, might believe he had legal grounds for redress from his employer. This is sheer nonsense if

on no other grounds than the fact that we cannot actually measure his significant dose with an error of less than $\pm$10–20 per cent. Yet it is difficult to evolve a law in this field where there is adequate latitude for ordinary reasonableness.

Consider the cost to plant operation if a legal limitation of 5 rems per year is put upon the individual worker. Plant management would be foolish to plan its operations in such a way that part or all of its workers would be allowed to take a dose closely approaching 5 rems per year. Experience has demonstrated that one or two people are bound to exceed this, and even though it is known that a slightly higher exposure is without harm, the employee might seek redress. Because of this fear, the plant must plan its operation so that, in general, no employee can receive more than say 40 or 50 per cent of the permissible exposure. This can be very costly and could seriously retard the atomic energy industry.

Where sufficient assurance can be provided that monitoring and recording methods are adequate, the widest latitude in the use of permissible exposure levels should be allowed. It is not known how this can be worked out legally but it seems fairly certain that in operations of any substantial size, it would pay the employer to have adequate measuring and recording systems, and then to be allowed wide latitude in his use of the recommended permissible levels.

The main difficulty may occur with the small users of radiation sources where there is transient employment and where the cost of monitoring equipment may be incommensurate with the cost of the operation. For such users, permissible exposures integrated over 1 week, or, at the most 3 months, would in all probability be the most feasible.

This problem may be overcome by means of a simple formulation relating a total permissible accumulated dose to the age of the individual. This interpretation says that at any age of 18 or over, the individual may be allowed to accept a dose such that his total accumulation in rems will not be in excess of 5 times the number of years over 18. The age of 18 is the youngest at which a person in the United States may be employed in radiation work. This age limit may be different in other countries.

The following is the exact statement by the NCRP:

Accumulated Dose. The maximum permissible accumulated dose, in rems, at any age, is equal to 5 times the number of years beyond age 18, provided no annual increment exceeds 15 rems. Thus the accumulated MPD = 5 $(N - 18)$ rems where N is the age and greater than 18. This applies to all critical organs except the skin, for which the value is double.

The interpretation given above implies that when a person starts radiation work for the first time, at some age over 18, he will be able to accept a higher dose per year provided that (1) it does not exceed 15 rems and (2) he does not exceed his age prorated maximum. This means essentially that the individual has built up a reserve of permissible exposure. But such a concept if exploited, carries certain obvious dangers. On the other hand, if every individual in an operation carries some reserve, it will unquestionably make for simpler and more economical protection procedures. The principle should not be exploited to the point where persons are required to accept radiation exposure that may with reasonable care be avoidable.

The age proration principle automatically holds down the exposure at the younger ages where it is most critical. It gives leeway at the older ages where it is less critical.

This principle places somewhat of a premium upon the employment of older persons with little or no radiation exposure history, expecially for tasks involving unavoidable exposure at levels above the average. However, the age prorated maximum is overridden by the yearly maximum which should not be exceeded. While an operator may never want to avail himself of the flexibility inherent in the age proration principle, its existence offers particular advantages as well as insurance in special circumstances.

Application of the principle is illustrated in Fig. 1. The area under curve A illustrates the region within which a person's accumulated dose must be contained under the new requirements. The broken curves drawn in this area show hypothetical illustrations of how the dose principle may be applied in individual cases. The maximum by age 60 would be 210 rems.

The region below curve B shows the comparable dose limits under the provisions that

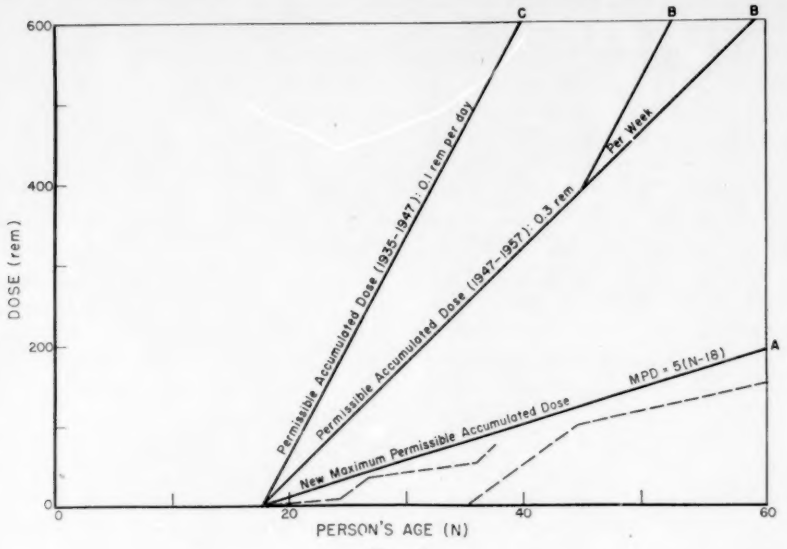


FIG. 1.

have applied in the U.S.A. from 1947 to 1957. Here at age 60 the maximum would have been 630 rems.*

The region under curve *C* shows the limits recommended in the U.S.A. between 1935 and 1947. The total at age 60 could have been 1260 rems.

For about the same period of 1934–1950 the maximum permissible accumulated dose as recommended by the International Commission on Radiological Protection could have been as high as 2520 rems by age 60—or more than 10 times the present new levels.

For design or planning or operational purposes, plants may wish to use average weekly, monthly or quarterly levels and to do their monitoring accordingly. Under our interpretation this will be permitted. Where it is desired to keep monitoring and recording procedures to a minimum, it would probably be wise, in many cases, to operate on a weekly or monthly basis anyway. The choice of procedure will depend largely upon (1) the economics of any particular situation and (2) the margin of uncertainty that would be introduced in the recorded accumulated dose of the individual.

Where a person's exposure in prior employment may be unknown or undocumented, he will be presumed to have had the maximum dose permitted up to his particular age.

It might be pointed out that the levels discussed above need not be modified by the addition of one emergency exposure of 25 rems during that person's lifetime.

Also it might be noted that these exposure levels are not modified by any radiation exposure received for medical reasons. On the other hand, it would be prudent for the employer to take any especially large medical exposures into consideration in the assignment of an individual to radiation work.

The discussions thus far have applied to irradiation of major portions of the body from external sources of radiation. They apply more specifically to the critical organs, which are the blood-forming organs of the trunk and long bones, the gonads, or the lenses of the eyes. An exception is the skin of the whole body when subjected to very soft radiation. For such conditions the dose limits may be doubled. Another exception would be the hands and forearms, and feet and ankles, for all of which the levels remain unchanged at the 1950 values.

* This does not include the possibility of doubling the weekly dose at age 45. If this is included the accumulated dose up to age 60 could be as high as 850 rems.

Permissible radiation levels for internal emitters have always been directly related to the external irradiation values, and this relationship will continue for occupational exposure. When the critical organ is the gonads or the whole body the MPD will conform to the age prorated limits. Correspondingly the MPD in air and water will, in these cases, be one-third the values heretofore specified for radiation workers. There is a fortunate circumstance here, in that for most radionuclides, the critical organs are highly localized and are other then the gonads or whole body. For such limited critical organs, the old concentrations will still apply—subject of course to correction as our knowledge increases. It is only for some 15 radionuclides that we must consider the whole body or, the gonadal dose. These would include such radionuclides as Na^{22}, Na^{24}, H^{3}, Cl^{36}, Cl^{37}, A^{37}, Br^{82}. It is thus clear that the new permissible dose levels will not too seriously affect our present practices with regard to most internal emitters.

Let us examine briefly some of the problems that will be introduced by the new protection standards. For radiation from external sources the new levels are effectively one-third of those in use up to now. This will not seriously affect the larger nuclear industries, except in relatively rare instances. Repeated warnings by the ICRP, and related national bodies, have indicated that future levels might have to be still lower. In any case the nuclear industry, by and large, has pioneered the development of conservative radiation practices, with the result that they have operated with a fairly large margin of safety. It has paid off. While that margin of safety will be narrowed, some margin will still remain. Also, with the flexibility inherent in the age proration principle, means will readily be found to comply with the new recommendations.

In the case of lighter industry, research, and medicine, the problems may in many instances, be more serious. In some situations, economic or structural considerations have led to practices which may have been more nearly marginal with respect to the old standards. In many instances also, it has been necessary to operate in close proximity to persons who are not classed as radiation workers. Under the old ICRP standards it was not permitted to expose non-radiation workers to levels greater than one-tenth of the occupational levels.

It has not been finally decided whether the non-occupational levels will be one-tenth of the new levels or remain at one-tenth of the old levels, but if we are to hold down adequately the average non-occupational dose for somatic as well as genetic reasons it may be necessary to work to one-tenth of the new standards. Since this is only 3 to 5 times the natural background radiation at sea level, it is obvious that serious protection problems have to be faced.

Consider, for example, the use of radioisotopes in medicine and industry where they are employed in telecurie therapy or radiographic examinations. Weight and mobility considerations alone dictate the lightest possible shielding container. Secondary protection afforded by walls and floors, if presently marginal, will have to be increased in thickness by about two half-value layers. This cannot only be costly in existing installations but in some instances may be completely ruled out because of floor-loading considerations.

There are two other problems not within the scope of this conference. One relates to the medical use of X-rays; since many such installations are in more or less conventional buildings it will be difficult to increase the shielding. Another problem involves high energy accelerators where two half-value layers of protective concrete might be as much as 30 cm.

In the isotope production and separation industry new shielding problems may arise in the secondary establishments. In the reactor installations there are probably already adequate margins of safety but this is less likely in areas where the materials are handled and fabricated into devices for the public.

As already mentioned, the new shielding problems in heavy nuclear industry will be slight. The standards for concentration of radioactive materials in air and water will not be changed for most elements. Problems of protection of the public will not increase greatly, mainly because most large installations are fairly well isolated.

To sum up the situation, it appears that because of the wise practices developed over the past 15 years, we have already prepared

ourselves against what might otherwise have been a severe dislocation of our budding nuclear industry. Continuance and extension of these wise policies will just as surely pay big dividends in the future. The elements of wisdom, common sense, and personal good judgment are essential if the uses of radioactive isotopes in the service of man are to be adequately exploited.

To give some concept of the nature of the overall protection problem it may help to show the relative contributions to man's radiation exposure from all major sources. Table 2 shows

Table 2. Radiation dose received up to age 30
(MPD = 14,000,000 man-rems per million persons)

	Dose (man-rems)
Natural radiation	4,000,000
Medical irradiation	5,000,000
Occupational exposure	150,000
Radiation in plant environs	
At one tenth of new MPD	(150,000)
At one-tenth of old MPD	450,000
Fallout	200,000
Total	9,800,000
Balance	4,200,000

the more important contributions to man's radiation exposure up to age 30. Most of the numbers are estimates which are probably on the high rather than the low side, and are based on conditions in the United States. The doses are expressed in man-rems per million of population. This is to emphasize that exposure of the population as a whole is the more critical figure.

The ICRP now recommends that the average per capita dose for the whole population should not exceed 10 rems up to age 30. This is in addition to the natural background of about 4 rems over the same period. Thus the total permissible dose would amount to 14,000,000 man-rems per million persons.

The dose from natural radiation acts over the full 30 years on the whole population and will contribute 4,000,000 man-rems.

Medical irradiation is averaged over the whole population for 30 years, and in the U.S.A. might possibly be as high as 5,000,000 man-rems.

Occupational exposure at an average of 5 rems/year acts on one-fourth of a per cent of the population over a period of 12 years, up to age 30, and contributes 150,000 man-rems.

Irradiation of non-radiation workers in the environs of radiation sources, might conceivably act on 1 per cent of the population for an average of 30 years, though this seems highly unlikely in the forseeable future. At a level of one-tenth of the new occupational levels this would contribute 150,000 man-rems. At one-tenth of the old levels the contribution would be 3 times as large, or 450,000 man-rems.

Fallout radiation would act on everyone for 30 years, and, at double the current rate, would contribute 200,000 man-rems.

Let us not be complacent about the seemingly low contribution to our total dose from occupational exposure. For the individual, it still amounts to 50 rems each 10 years, and this is over 5 times higher than the average per capita dose for the whole population. It is however, to be accepted like any other occupational risk in industry—if one is unwilling to accept the risk, one seeks some other occupation. After all, one does not have to be a steeple-jack or a coal miner—or a radiation worker.

The total from all sources is 9,800,000 man-rems, leaving a balance of roughly 4,000,000 man-rems. Some of this balance will be consumed by the use of radioactive household devices, nuclear power, etc. However, it should not be used up simply because it appears to be there. Bear in mind, that we must still operate on the principle that no radiation taken by man unnecessarily is good for him. Rather than try to use up our radiation reserve, we should endeavor, by all means possible, to increase it. With the pressures that will be upon us to exploit new uses for radioactive materials this will not be easy. But resist these pressures we must—within bounds of common sense and wisdom—if we are to continue to receive the bounties of our new servant.

Health Physics Pergamon Press 1958. Vol. 1. pp. 71–75

HEALTH PHYSICS RESPONSIBILITIES TO MANAGEMENT*†

G. HOYT WHIPPLE‡
University of Michigan, School of Public Health, Ann Arbor, Michigan

(*Received* 21 *November* 1957)

Abstract—Health physics is a relatively new and rapidly expanding profession. As a consequence its aims and responsibilities are neither well-defined nor generally understood. With the frank intent of promoting discussion, this paper outlines the kinds and degrees of services that a health physicist may be called upon to render to the management of a company that is embarking upon radioactive work for the first time. A health physicist in such a company has four principal areas of responsibility: the safety and health of plant employees, the safety and health of the public, adequate legal basis for protection, and an economical operation. The general manner in which these responsibilities are met in the design, development and operation of a plant is described, and an attempt is made to show how critically the evolution of health physics as a respected profession depends at this stage on the performance of the individual practitioner.

It is characteristic of a new and rapidly expanding field that its place in the professional community is poorly defined. Today, any attempt to draw sharp boundaries for the field of health physics would be not only futile, but ill advised. However, from time to time it is desirable to examine critically the standing of the field, its opportunities and responsibilities, and the probable directions of its development. With the intent of promoting discussion, I shall outline the services that a health physicist may be called upon to render to the management of an enterprise that is about to utilize radioactive materials for the first time. The comment and criticism which I hope this outline will elicit will be of value both in the design of suitable health physics training programs and in the certification of qualified persons in the field.

The responsibilities of an industrial health physicist to his management involve a number of complex obligations which I shall describe briefly under four general headings:

(1) *Employees.* The health physicist must see that the plant employees have working conditions, instruments and assistance which will ensure radiological safety in the performance of their duties. Further, he must assist them to develop an intelligent appreciation for the nature and control of radiation hazards, and to this end he must be prepared to discuss with them honestly and *intelligibly* any fears or doubts they may have about their work.

(2) *Public.* The health physicist must assure by all possible means that the operations in the plant involving radiation or radioactive materials do not endanger the health and safety of persons living near it, and he must be in a position to provide all the information that these people require to decide for themselves that these operations do not, in fact, represent a threat to their well-being.

(3) *Legal.* The health physicist must see that the plant operation conforms to all pertinent federal, state and local regulations, and beyond this, he must do what he can to promote a

* This paper is based in part on material presented by the author in a talk, *Qualifications on Professionals in the Field of Radiation Hazard Control*, given at the Atomic Industrial Forum meeting on Utilization of Nuclear Engineering Standards by State Governments, New York City, 6 February 1957, and also on two articles on *Radiation Safety in the Nuclear Industry*, prepared by Dr. W. D. Claus and the author for the Whaley-Eaton Service, Atoms for Peace, Washington, D.C., 12 and 19 April 1957.

† Presented at the Health Physics Society Meeting, Pittsburg, Pennsylvania, 17 June 1957.

‡ Health Physics Consultant to Atomic Power Development Associates, Inc., Detroit, Michigan.

spirit of intelligent co-operation with the various public health officials. He must protect the company against claims of negligence and against spurious claims arising from the operation of the plant.

(4) *Economics.* Finally, this paragon of skill and tact must accomplish all these things in the most economical manner possible.

POLICY

The first step in establishing radiation control is the formulation of a satisfactory policy statement to be issued by management. Such a statement will serve several purposes:

(1) It will acquaint the management with the significance of radiation hazards and with the regulations for their control.

(2) It will provide the health physics organization with an explicit statement of the support it can expect from management.

(3) It will save considerable friction as the design and operational features of the plant are worked out.

(4) And, finally, it will serve as a powerful tool for winning public acceptance of the undertaking.

A crucial factor in the policy statement is the practical application of permissible exposure levels. Maximum permissible exposure levels for workers and for the public are, of course, specified in many cases by law. However, the extent to which it is desirable to keep below the prescribed maxima is a matter worthy of careful deliberation. Management will quite properly be conscious of the cost of providing more protection than is required by law. The health physicist will be called upon to point out that standard operating procedures must be planned for an *average* exposure well below the maximum permissible, if the maximum is not to be exceeded, that extra precautions will contribute to better relations with employees, organized labor and the public, that liability for radiation damage to health can be reduced by an exposure policy more stringent than the law requires, and that the present state of knowledge of long-term biological effects of radiation makes it prudent to keep all exposure to the lowest practicable level.

Hazards report

Many nuclear facilities, and all reactor plants are required to submit a hazards summary report to the AEC as a prerequisite for a license. The preparation of this report involves critical evaluation of all the potential hazards to those working in the plant and to the public, under both normal operating conditions and all forseeable accident conditions, and of the steps that are taken to reduce these hazards. The health physicist must obviously be intimately familiar with the workings of the contemplated plant if he is to make his full contribution to this study.

Design review

In the design of the plant itself the health physicist has a unique opportunity to see that radiological safety is incorporated throughout the planning. To the extent that he is able to indoctrinate engineers and architects in the importance of considering radiation hazards and of eliminating or reducing them, the designing-in of safety will be simplified and made more effective. The health physicist can give invaluable service at this stage by pointing out unsuspected difficulties and suggesting solutions to them. Before the design and operational aspects of the plant can be considered complete, they must be evaluated for radiation and other hazards. Of particular importance in this connection are:

(1) Ventilation and gas systems.
(2) Shielding.
(3) Plant layout and traffic patterns.
(4) Contamination control points.
(5) Waste management systems.
(6) "Hot" facilities.
(7) Maintenance and repair in high level radiation areas.
(8) Shipping arrangements for radioactive materials.

This review can be expected to result in a plant that is safer and more economical to operate than a plant designed with only casual attention to radiation hazards. The costs of well-conceived safety measures will be reclaimed many times over by savings in operating costs.

Site selection and environmental surveys

Concurrent with design of the plant is the selection of a suitable site. Considerations of radiation safety are among the factors that must be weighed in choosing a site for a nuclear installation. Gaseous and liquid effluents, waste disposal, potential consequences of accidents—all involve attention by the health physicist.

After the plant site has been selected, it will be subjected to a pre-operational environmental survey. Nuclear plants may affect the environment in which they operate. Moreover, a primary public responsibility of the management of such a plant is to assure that environmental effects are held well below levels of any significance to health and safety. Therefore, it is essential to have a measure of the levels of radiation and radioactive materials that prevail in the environment of the plant before the facilities go into operation. Such a survey has three objectives:

(1) It will make possible the prompt recognition of any increase in environmental levels related to plant operation so that the causes may be found and corrected long before levels of any significance to health and safety. Therefore, it is essential to have a measure of the levels of radiation and radioactive materials that prevail in the environment of the plant before levels of any significance to health are reached.

(2) It will reduce the number of false alarms resulting from unusually high radiation levels, natural or artificial, which may be unrelated to plant operation, to the minimum consistent with the first objective.

(3) It will protect the plant from the claim of negligence and from claims for injury or damage unrelated to plant operation.

The equipment and methods used in the pre-operational environmental survey will also be used in large part for the further environmental surveys that will continue for the life of the plant.

Operating procedures

The development and planning of standard operating procedures must include detailed appreciation for the existence and control of radiation hazards. Many of the customary routines for non-radiation inspection, maintenance and repair need modification to achieve minimum radiation exposure consistent with efficient operation. Some routines, normal in conventional plants, will have to be accomplished by remote means, or under carefully regulated "special work permit" conditions.

Manual. Standard operating procedures concerned specifically with control of radiation hazards must be developed and kept current with the best practice. Furthermore, these procedures must be available in detailed written form if management is to avoid the charge of negligence in the event of litigation, or possible loss of license. Methods of record keeping must be worked out in considerable detail for such things as: decontamination, particularly of people, areas and shipments; instrument calibration; radiation and contamination surveys; inventory of radioactive materials; evaluation and control of radiation exposure.

Instruments. The selection of instruments suitable for the evaluation and control of radiation hazards in any particular plant is still not any easy undertaking. A good deal of judgment is required not only to select the types and quantities of instruments needed initially, but also to make the modification, substitution and elimination necessary to keep the plant in step with the best standard practice.

Eventually, when the ferment of new developments subsides, instrumentation for radiation control will become only slightly more complex than that for any of the familiar process variables; temperature, pressure, flow etc. But at present and for several years to come it is a matter demanding the careful judgment of a trained health physicist.

Monitoring routines. The proper location and use of radiation instruments are as important as their selection. Fixed, continuous radiation monitors can provide automatically much information that might otherwise have to be obtained with portable instruments in the hands of skilled surveyors, thereby saving expensive man-hours and unnecessary personnel exposure.

Monitoring devices for airborne radioactive materials are even more important than those

for radiation. The low levels of the maximum permissible concentrations and the presence of natural radioactive materials in the air place extraordinary demands on these devices. It is not surprising that instrumentation in this area often falls short of operating demands. With available instrumentation for airborne radioactivity, careful judgment is necessary if safe working conditions are to be maintained without unnecessary restrictions.

Not all kinds of data in all areas, however, can be provided by fixed continuous monitoring instruments. Sometimes, suitable coverage, even of routine type industrial operations, is obtainable only through well-planned surveys by trained monitors, making spot samples and observations with mobile and portable instruments. Many operations, laboratory and maintenance, for example, are non-routine in nature, and their safe conduct requires custom-made surveys and evaluations. Such surveys will fill in the unavoidable gaps in the network of fixed monitors. They also will provide data that, taken with information from fixed monitors, will characterize occupational and public environment in detail as to the existence of real and potential hazards in both space and time.

Personal exposure from sources of radiation external to the individual's body is measured by the familiar photographic film badge and/or by pocket ionization chambers or dosimeters. Exposure from sources of radiation inside the individual's body is estimated from analyses of the radioactivity in the excreta, or from measurements of the very small amounts of radiation emitted from the individual's body.

The essence of all these measurements is that they be carried out faithfully according to detailed procedures. Moreover, the procedures must be subject to rigid control, and the data and the interpretation placed on them must become a matter of permanent record.

In addition to checking on plant control, personnel monitoring serves as a measure of worker training and co-operation. It thereby constitutes the primary record of every employee's occupational radiation exposure history as required by federal and many state regulations.

Records

A health physics organization has only two things to show for its efforts: a safe plant and a massive collection of records. These two products are of about equal importance. The value of a complete record of procedures, instruments, calibrations, data, calculations, interpretations, and final evaluations for analyzing the degree of protection achieved, plus the *cost* of this protection, and for indicating improvements, is fairly obvious. The value of this record in the event of litigation or threatened loss of license may not be so obvious and deserves more explanation.

The general opinion is that in the event of an injury or damage attributable to radiation, the doctrine of *res ipsa loquitur* (it speaks for itself) will apply. The burden of proof will rest with the plant operator to show that he is not guilty of negligence and that radiation hazards have been under competent control. If the operator cannot show this, he will be deemed negligent and will be held liable for any loss. For this reason insurance companies can be expected to exert pressure for systems of monitoring control and records sufficient to withstand legal tests for negligence.

Educational activities

For workers. The most important single factor in any safety program is the intelligent cooperation of the workers and their immediate supervisors. Since radiation is a hazard that cannot be seen, felt or tasted, and one that does not produce an immediate obvious injury, the importance of intelligent cooperation is even greater than with conventional hazards.

The workers cannot be protected and cannot protect themselves if they are ignorant of the hazards. The most reasonable way to gain the co-operation of people working in plants is to introduce them to the fundamentals of radiation, the biological damage caused by radiation, the amounts of radiation significant for biological damage, and the measurement and control of radiation. Every individual in the plant should be (and in fact may be required by law to be) indoctrinated with at least an elementary understanding of these matters. As a minimum, discussions should be carried to the point where

every individual is reasonably able to satisfy his own doubts and questions.

Effective indoctrination and the subsequent discussion is not a simple thing. Unless it is conducted by competent people, there will remain doubts in the minds of some, first as to the hazards of their work, and second as to the ability of the health physics group to cope with the hazards. Without the confidence and respect of workers, supervision and management, the health physics effort is a futile one.

For the public. The widespread public interest and concern in the dangers of atomic energy, and the fundamental right of people to know about any matters that may affect their well-being, make it imperative that management provide the public, the press and health officials with such information on hazards and safeguards as they may wish.

Thorough and frank discussion of the technical and health aspects of an atomic plant carried on with state and local officials throughout the early planning stages will go a long way toward gaining the co-operation of these key people and winning public acceptance of the plant. For the next few years at least, such discussions will constitute an effort in educating public officials in new fields. To the degree that this educational effort is carried out thoroughly and effectively it will pay large dividends in the informed cooperation it will win. This effort will demand the assistance of the best scientific and engineering talent that management can provide. It is obvious that the health physicist has an essential role to play in this effort.

Emergency plans

The evolution of a realistic plan for coping with any emergency condition that might develop within the plant is one of the major items on which management will have occasion to work with local health and safety agencies. Aside from the obvious benefits of a sound emergency plan, this work will be of public relations and legal value as a manifestation of the degree of responsibility management takes. It also will be of potential value in the event of community emergencies that in no way involve the plant.

The common ground between the plant emergency plan and local civil defense planning is an important and valuable place for co-operation. The plant health physics organization and equipment comprise a valuable asset to the civil defense structure in the community. When properly developed, both groups will find mutual benefit from the association.

Research

Sooner or later in the operation of almost every plant problems will arise that cannot be solved by routine instruments and routine methods. Where time and competence permit the solution of such problems by the health physics group in the plant, such investigations will stimulate and broaden the understanding of the group. If at all possible, health physicists should be encouraged to carry out research and development investigations suggested by the plant operation. Such investigations contribute both to the growth of the individual and to that of the field.

Certain problems will arise that lie outside the capabilities of the plant health physics group to solve. These should be referred to a qualified laboratory for solution. National laboratories, universities and commercial laboratories can usually assist in carrying through the necessary investigations, once the importance of the problem has been brought to their attention.

SUMMARY

I have attempted to show in general outline the types of responsibilities that a health physicist has to the management of a concern that is undertaking to use radioactive materials for the first time. Even in outline form, these are impressive responsibilities that demand profound breadth and depth of training and experience. The fact that today many of these responsibilities are, by their very nature, ill-defined, does not in any way detract from their importance.

It is the hope that this paper, and the discussions it may arouse, will help us to decide where we are, and how best to proceed in the exciting years immediately ahead.

Health Physics Pergamon Press 1958. Vol. 1. pp. 76–84

EXPOSURE CRITERIA FOR EVALUATING THE PUBLIC CONSEQUENCES OF CATASTROPHIC ACCIDENTS IN LARGE NUCLEAR PLANTS*

F. P. COWAN and J. B. H. KUPER
Brookhaven National Laboratory

(*Received* 19 *November* 1957)

Abstract—Various types of direct exposure to the fission product cloud from a catastrophic reactor accident are analyzed and expressed in terms of an acceptable emergency dose (AED) due to 1 c-sec/m^3 of exposure. The problem of combining these exposures is considered and criteria are suggested for four categories of personal injury. Indirect exposure due to deposition of fission products is also considered and limits are suggested to be associated with evacuation and varying degrees of restrictions on the use of land.

This paper is a condensation of a more detailed development of exposure criteria included in a study of the consequences of catastrophic reactor accidents carried out by a group at Brookhaven National Laboratory. A very brief summary of the methods and results for the whole study is given in an appendix.

DURING late 1956 and early 1957 a group at Brookhaven National Laboratory carried out a study of the public consequences of catastrophic accidents in large nuclear power plants. The authors of this paper were responsible for suggesting criteria for internal and external exposures to be associated with varying degrees of personal injury and for estimating the levels of fallout to be associated with various restrictions on the use of land or buildings. While the considerations leading to our choice of exposure and fallout criteria are the major concern of this article, a summary of the findings of the whole group[1] has been included in Appendix I since they will be of considerable interest to health physicists.

Two main types of exposure were considered. The first was direct exposure due to immersion of an individual in the cloud of radioactivity or due to subsequent irradiation by fission products taken into the body from the cloud. The second was exposure due to the presence of fission products deposited from the cloud or due to such deposited activity finding its way into the body. Criteria based on these two types of exposure were developed and will now be somewhat briefly discussed.

DIRECT EXPOSURE

Introduction

Estimation of the consequences of exposure to a cloud of fission products is a difficult task. In fact, the conclusions reached will be only approximate since the direct effects on humans of exposures of this character are largely unknown. However, even the admittedly rough exposure criteria that result are useful and indeed are prerequisite to the evaluation of costs connected with reactor catastrophes.

We will use four categories in classifying exposures resulting from immersion in the radioactive cloud. Table 1 shows the physiological effects associated with these categories, the whole body γ dose assumed to correspond to these effects and the values of exposure now to be deduced. It will be noted that the unit chosen for expressing exposure is the curie-second per cubic meter (c-sec/m^3), i.e. the product of cloud concentration and time spent in the cloud.

Direct exposure is not only due to whole body

* Paper presented at the 1957 meeting of the Health Physics Society. Work done under the auspices of the Atomic Energy Commission.

Table 1. *Exposure categories and values*

Category		Full F.P. release (c-sec/m³)	Volatile F.P. release (c-sec/m³)
A	Lethal exposure	>400	>350
B	Illness likely	400–90	350–80
C	Injury unlikely, but some expense may be incurred	90–10	80–10
D	No injury or expense	<10	<10

β and γ exposure during passage of the cloud, but also to the inhaled fission products. Thus we are faced with the necessity for combining several components of exposure. In doing this we have made use of a quantity called the Acceptable Emergency Dose (AED). This is defined as 25 r of whole body γ exposure or its equivalent of other types of exposure. Such a dose is considered to be acceptable from the point of view of this study in that no injury or expense is anticipated at or below the value chosen. Acceptable emergency dosages for rescue operations, where greater risks or actual damage to personnel are necessary, would of course be considerably higher. The procedure used in the following paragraphs is to work out the number of AED corresponding to 1 c-sec/m³ exposure for each of the major components of exposure. These are then combined and limits set for the four exposure categories. Only an outline of the computations can be given here and the reader is referred to the full report for details.[1]

The fission product mixture assumed in making these computations, was one characteristic of a 500 thermal megawatt reactor that has operated for 180 days. The chief components of this mixture are shown in Table 2 below. As indicated in Appendix I limits were required for two types of accidents, a release of all volatile fission products plus 1 per cent of the strontium, and the release of 50 per cent of all fission products. The former will be referred to as the volatile case, and the latter as the fission product case.

WHOLE BODY γ AND β EXPOSURE

Computations of whole body γ and β exposures were based on immersion in a semi-infinite cloud. It was expedient to express all exposures to the cloud in c-sec/m³ as of 24 hr after the incident. Thus, in deducing the dose corresponding to 1 c-sec/m³ of exposure it was necessary to correct for decay since the time of cloud passage which was taken as 2 hr after the accident. This is an intermediate value which will be too late for close-in areas but too early for the populous areas farther away. Because of the slow decay, dose calculations for the fission product release are not very sensitive to the choice of arrival time. In the volatile case, the estimated limits depend significantly on the arrival time and a more sophisticated analytical treatment would be desirable.

In correcting for decay, a $t^{-0.2}$ decay law was used for the fission product case. In the volatile case it was found empirically that total β or γ energy would decay as $t^{-0.8}$ in the time interval from 2 to 500 hr after the accident and this law was utilized. It is recognized that a considerable degree of approximation is involved in these assumptions, but they are considered to be justifiable in view of the other large uncertainties involved in the study as a whole.

Using 0.7 MeV as the average γ energy, an exposure of 1 c-sec/m³ was found to correspond to a whole body γ exposure of 0.28 rad (0.0112 AED) for the volatile release and 1.26 rads (0.0505 AED) for the fission product release. Since the cloud will not be semi-infinite in

Table 2. *Fission product inventory as of* $t = 24$ hr

	(c)
Total fission products	4.1×10^8
Strontium-90	3.8×10^5
Strontium-89	1.7×10^7
Cesium-144	8×10^6
Plutonium-239	3.8×10^3
Iodines	5×10^7
Noble gases	3.4×10^7
Total volatile fission products	8.4×10^7

extent, doses while in the cloud are overestimated, but this is counterbalanced to an undetermined extent by some dosage received when the cloud is aloft and there is no exposure in c-sec/m^3.

Whole body beta exposure from the passing cloud would be less than the gamma exposure and, since it affects principally the skin, should not contribute significantly to the acceptable emergency dose. However, it should be borne in mind that beta dosage may become important if material is deposited on the body and not washed off promptly.

Beta lung dosage. The lung dose will be predominantly due to beta radiation. It was computed, using 25 per cent lung retention of particulates and the decay laws mentioned above. A value of 0.4 MeV was used as the average β particle energy. Thus for the fission product case, 1 c-sec/m^3 of exposure resulted in lung retention of 55 μc as of $t = 24$ hr and a dose of 1.21 rads during the first 24 hr. Dosage for the balance of the first week was estimated to be about twice this value. For the volatile case, lung retention was only 22 μc, since 40 per cent of the activity was assumed to be in the form of noble gases, and dose was 0.9 rad.

There is no firm basis on which to convert lung dose to equivalent whole body gamma dose. However, it is noted that the recommendations of Handbook 59, United States Bureau of Standards, permit five times as much beta exposure to the basal layer of the skin (whole body) as gamma and also permit five times as much exposure to the extremities as to the whole body. It seems quite conservative to use a factor of 5 in reducing lung exposures to whole-body equivalents. On this basis, 1 c-sec/m^3 will result in 0.24 r equivalent (0.0096 AED) for the F.P. case and 0.18 r equivalent (0.0072 AED) for the volatile case.

Strontium-90 *bone dose.* Some of the fission products deposited in the lungs will of course find their way to other parts of the body and result in other components of dose. A value of 1 μc firmly deposited in the bones as a result of the assumed accident was taken as 1 AED. This amount corresponds to an average of 0.21 μc over the next 40 years, if we use the effective half-life of 2200 days, or one-fifth of the permissible occupation body burden. It was assumed that only 10% of the inhaled Sr was retained fairly permanently in the bones so that inhalation of 10 μc was taken as corresponding to an AED. A dose of 1 c-sec/m^3 would thus be 0.02 AED as far as Sr^{90} is concerned in the F.P. case. Since Sr is depleted by a factor of 20 in the volatile case, the value there would be 0.001 AED.

Strontium-89 *bone dose.* The analysis for Sr^{89} was tied to that for Sr^{90} on the basis that the relative hazards from these two isotopes should be proportional to the product of average particle energy and effective half-life. On this basis seventy-seven times as much inhaled Sr^{89} as Sr^{90} would correspond to 1 AED or 770 μc. Assuming 22 per cent deposition in the bones this would give an infinity bone dose of about 50 r which seems like a reasonable limit for a single emergency exposure of this nature. The Sr^{89} inhaled in an exposure of 1 c-sec/m^3 will amount to 9 μc (0.0116 AED) for the F.P. case and 0.45 μc (0.00058 AED) for the volatile case.

Cesium-144, *Praesodymium*-144. Using the same line of reasoning as for Sr^{89}, it would appear that the permissible emergency deposition of Ce^{144} would be 0.12 times that for Sr^{89}. However, transfer to the bones is less efficient and Handbook 52 gives 10 per cent rather than 22 per cent as the fraction of the inhaled amount that is deposited there. Thus inhalation of 203 μc of Ce^{144} should correspond to 1 AED. A 1 c-sec/m^3 exposure involves inhalation of 4.3 μc of Ce^{144} or 0.021 AED for the F.P. case. For the volatile case cesium would be negligible.

Plutonium-239 *lung dose.* For the assumed reactor, Pu^{239} would constitute only 0.00093 per cent of the total activity. An estimate of the dosage corresponding to 1 c-sec/m^3 of exposure was made using data from Handbook 52. It was assumed that the plutonium would be in insoluble form so that the lungs would be the critical organ. A dose of 0.04 μc retained in the lungs or five times the occupational limit was taken as the AED since the biological half-life is given as only 1 year. Pu^{239} appears to contribute little to the dose, 1 c-sec/m^3 corresponding to only 0.0012 AED for the F.P. case and to a negligible value for the volatile release. For soluble Pu^{239} the same result is obtained since

0.04 μc is still an appropriate choice of body burden.

Iodines, thyroid dose. The iodines present after 24 hr of cooling are chiefly I^{131}, I^{133} and I^{135}. The critical organ is the thyroid which appears to be relatively radioresistant. A dose of 25,000 to 30,000 rads is estimated to result within 6 months in sufficient symptomatology in one-sixth to a half of the persons so irradiated to cause them to seek medical care for apparent deterioration in their well-being. There are, however, cases of thyroid malignancy in adolescents who are believed to have received doses of the order of 200 r to the thyroid incidental to irradiation of the thymus or adenoids as young children. Such cases are fortunately very rare, although large numbers of children received irradiation in the 1930's and 1940's. It seems reasonable to set the AED at one-tenth of the lowest figure at which symptoms may be expected, or 2000 rads to the gland, although the possibility of an occasional tumor at this level must be admitted.

A value of 400 μc of the iodine mixture in the thyroid as of $t = 24$ hr was roughly established as giving the above dose. Using 15 per cent as the fraction of inhaled iodine reaching the gland, we find that 2660 μc inhaled corresponds to 1 AED. Exposure to 1 c-sec/m^3 would involve inhalation of 24 μc or 0.01 AED for the full F.P. release and 0.049 AED for the volatile release, where the iodines amount to 40 per cent of the total activity.

G.I. tract dose. Estimation of the dose to the intestines or other organs does not appear feasible even in the crude fashion used above for the lung, bone, and thyroid doses. This is partly because most of the activity would have been swallowed after removal from the bronchi by ciliary action, a process which would continue for days. As a rough guess, the amount of material going through the intestinal tract would be the same as the estimated amount remaining in the lungs. Assume that passage requires 24 hr and that the tract weighs 2 kg. In the fission product release 1 c-sec/m^3 is considered to leave 55 μc in the lungs. This amount of activity in the intestines would give a dose of 0.56 rad. Although the intestines are, generally speaking, rather radiosensitive, a single dose of 50 rad would probably be acceptable. Therefore 1 c-sec/m^3 is taken to give $0.56/50 = 0.0112$ AED to the intestinal tract. The volatile release results in practically no activity except from the noble gases and iodines, therefore the dose to the intestines would be negligible.

Total direct exposure. At this point it becomes necessary to sum up these various effects. Almost no data exist on the additivity of partial body exposures. In the case of the limit for no injury (category *D*) it is conservative simply to add up the partial exposures as expressed in AED, but this will not work for the sickness or lethal limits. One cannot half kill a man by hanging and half kill him by shooting and end up with one dead man.

The effects of 1 c-sec/m^3 of exposure expressed in AED units are recapitulated in Table 3. It seems safe to pick an exposure of 10 c-sec/m^3 as the limit for category *D* in the full fission product release. In the volatile case it appears that practically nothing counts except the external gamma dose and the dose to the thyroid. There is no reason to suppose that these effects are synergistic; in fact the thyroid appears to have no particular relation to radiation sickness. Thus it seems conservative to pick the same figure, 10 c-sec/m^3, for the category *D* limit in this case too.

The situation becomes much less clear when

Table 3. *Effects of* 1 c-sec/m^3 *of exposure expressed in AED*

	Full F.P. release	Volatile F.P. release
External γ dose	0.0112	0.0505
Lung β dose	0.0096	0.0072
Bone dose from Sr^{90}	0.02	0.001
Bone dose from Sr^{89}	0.0116	0.00058
Bone dose from Ce^{144} + Pr^{144}	0.021	—
Lung dose from Pu	0.0012	—
Thyroid dose	0.01	0.049
G.I. tract dose	0.0112	—
Total	0.096	0.108

an attempt is made to pick the lethal dose (the limit for category *A*). Probably the bone seekers may be added together, although the long time over which the Sr^{90} delivers its dose makes this somewhat doubtful. Also, the lung beta dose and the Pu dose may possibly be additive. An exposure of 400 c-sec/m^3 would then give a dose of 112 r whole-body, 540 rads to the lungs in the first day or 1620 rads in the first week, 942 rads to the bones over a few years and 224 rads to the intestinal tract. In addition there would be an undetermined amount of radiation to the blood and other organs, and 8000 rads to the thyroid, which would not be considered as contributing to morbidity. Whether this combination of insults would be sufficient to cause death is certainly not known, but it does not seem unlikely. Therefore, a figure of 400 c-sec/m^3 will be adopted as the lethal limit for the fission product release.

In the hypothetical volatile release the situation is much less complicated. The lung beta dose will result principally from iodines on the way to the thyroid, therefore probably only the first day need be considered. The thyroid dose will be neglected since even complete destruction of the gland is not lethal. An exposure to 350 c-sec/m^3 would result in 450 r whole body plus 325 rads to the lungs and 28 rads to the bones. This exposure seems a good choice for the category *A* limit, since the lung and bone exposures are trivial compared to the whole-body gamma dose.

Selection of a figure for the boundary between categories *B* and *C* is fraught with uncertainty because of the varying susceptibilities and possible synergic effects. In principle, of course, a tumor might be induced by a very small dose, but the chances are small. Perhaps the best way to proceed is by simple extrapolation. If 450 r whole-body is the mean lethal dose and 100 r may produce illness, it seems reasonable to suppose that $(100/450) \times 350 = 78$ c-sec/m^3, rounded off to 80 c-sec/m^3, would be the illness limit for the assumed volatile release. The same procedure gives a figure of about 90 c-sec/m^3 for the illness limit in the assumed full fission product release, but this figure is to be regarded as much less reliable than the corresponding one for the volatile release.

Table 4. Ranges for surface deposition of activity

	Range	Full F.P. release (c/m^2)	Volatile F.P. release (c/m^2)
I	Urgent evacuation (within 12 hr) necessary	>0.2	>0.1*
II	Evacuation necessary	$>10^{-2}$	>0.1
III	Severe restrictions on land use, possible temporary evacuation, restrictions on outdoor work.	10^{-2}–10^{-3}	0.1–10^{-2}
IV	Probable destruction of standing crops, restrictions on agriculture for first year	10^{-3}–10^{-4}	10^{-2}–10^{-3}
V	No expense likely	$<10^{-4}$	$<10^{-3}$

* Unless adequate shelter is available.

INDIRECT EXPOSURE

Introduction

In dealing with the problem of deposition of fission products we have recognized five situations as shown in Table 4. As in the case of direct exposure, setting up criteria is difficult. Comparatively little is known about the problems involved in living in an environment heavily contaminated by radioactive deposition. It is obvious that external whole-body gamma irradiation would be significant at sufficiently high contamination levels, and that even at low levels problems would exist in agriculture, particularly dairy farming. Very little can be said, on the other hand, about the intermediate situation. The extent to which dust deposited on the landscape would find its way into food and water supplies is not known. However, rough limits can be set by considering several of the major factors in detail on the basis of available data, combined with reasonable assumptions. The curie per square meter (c/m^2) will be used as the unit for characterizing deposited activity.

Whole body gamma and beta dose from fallout.

Using 0.7 MeV as the average gamma energy, data from *The Effects of Atomic Weapons*, and making allowance for build-up due to scattering and reduction due to small ground irregularities, it was estimated that the gamma dose rate 3 ft above ground contaminated with 1 c/m^2 of gamma activity would be 10.6 r/hr. With the previously used assumption that the cloud would pass over 2 hr after an accident, the integrated dose for the first day due to 1 c/m^2 would be 226 r for the fission product case and 520 r in the volatile case. The dose rates at $t = 2$ hr due to this fallout would be 17.4 r/hr and 77.5 r/hr respectively. The total dose for the first 90 days was determined for the fission product case to be 11,500 r. On the ninetieth day the fission product activity will have decayed to only 40 per cent of the value at the end of the first day. For the volatile case only strontium and some long-lived daughters of the noble gases would remain after 90 days.

It should be noted that the figures in the previous paragraph apply only to an individual who spends all his time outdoors on flat terrain. For most people, except outdoor workers, it would seem reasonable to allow a factor of 5 for ground irregularities and the shielding effect of buildings.

Whole body beta exposure has been neglected on the presumption that the individuals would wash and change clothes promptly after the fallout. In a few cases in ranges I and II, particularly when coupled with direct exposures in category *B*, skin burns caused by beta radiation might be a contributing factor to the overall injury.

Evacuation limits. A dose rate such that 25 r would be received in the first 12 hr would appear to call for urgent evacuation and the range *I* limits shown in Table 4 have been set on this basis. In the case of the volatile release, the dose rate drops off quite rapidly. Thus, if good shelter is available, it would be better to wait a day before evacuating.

The limit for range II was set for the volatile case by stipulating a 3 month dose of 50 r, making no allowance for decontamination effected by the weather. In the fission product case only 25 r was used since the dose in the first year would amount to about 75 r. Actually, under most circumstances weathering would result in a considerably lower dose.

Range III represents a situation where evacuation would not be necessary as long as there were severe restrictions on living habits. For instance, long hours of outdoor work particularly in dusty atmospheres would not be permitted, people living in trailers or flimsy dwellings might have to be relocated and some activities such as dairy farming might be temporarily or permanently discontinued. Obviously, the uncertainties are such that specific restrictions would have to be based on measurements made on the spot after the accident.

Effects on agriculture. The occupational limit for Sr^{90} in humans is 1 μc corresponding to about 1 μc/kg of calcium. It is customary to require a reduction to one-tenth of this value for non-occupational exposures. However, there is considerable discrimination against Sr as compared with Ca in the pasture-cow-milk transfer and it seems reasonable to use 1 μc of Sr^{90} per kg of calcium in the soil as the maximum permissible concentration (1 MPC).

For the postulated reactor, 1 MPC of Sr^{90} in the top $2\frac{1}{2}$ in. of average soil would correspond to a total of 2.8×10^{-4} c/m^2 fission products deposited as of 24 hr after the release. After the first year, when the Sr^{89} and most of the other activities would have decayed, the Sr^{90} would be the limit on use of land, particularly for dairy farming. If it is assumed that the deposited material could be diluted with calcium by plowing to a depth of about 10 in., and that substandard soils could be brought up to normal by application of lime, it appears that land contaminated to the extent of 10^{-3} c/m^2 of fission products could be returned to production after a year. Land more heavily contaminated than this probably could not be used for dairying for a very long time, but other types of farming might be permitted. This figure of 10^{-3} c/m^2 happens to be the same as that suggested above for the lower limit of range III on the basis of external exposure. The agreement is fortuitous.

It seems reasonable to pick a figure of 10^{-4} c/m^2 for the lower limit on range IV although there is very little, if any safety factor here. At this level the Sr^{89} would be about ten times the

long-term MPC calculated on the occupational exposure limit, and many other isotopes would appear in crops. It is probable that anything raised on land contaminated to more than this extent would be barred from distribution for human consumption.

For the volatile release the strontiums would be depleted by a factor of 20. There might, however, be some troublesome daughters from the noble gases. It seems reasonable to allow ten times as much contamination here as in the fission product case and to set the lower limit for range III at 10^{-2} c/m^2 as suggested above. Correspondingly, the lower limit for range IV would be 10^{-3} c/m^2. Here it might not be necessary to lose a whole year's production as in the fission product release. Only if a crop were well along toward harvesting at the time of an accident would it be necessary to destroy it.

In range V, below 10^{-4} and 10^{-3} c/m^2 respectively, there probably would be no need for restrictions, although anyone with a Geiger counter could demonstrate the presence of radioactivity. Of course, vegetables and other food crops grown on this land should be thoroughly washed before eating. The fact that most people in the U.S.A. get their vegetables and fruits from widely separated places would be a very helpful factor in reducing the intake of activity. A family in a contaminated area trying to be self-sufficient by growing most of its own food and keeping a cow might ingest somewhat more activity than intended during the first year even at levels of contamination below the range IV limits.

DISCUSSION

Appropriate action by local authorities and the people in the area would cut the number of exposures in categories *A* to *G* very markedly. If the affected population could be warned to get into shelter with the windows closed while the cloud passes, there should be negligible inhalation of radioactivity and a great reduction in the gamma dose. Warning would probably not come in time to help those in the immediate vicinity of the reactor but should be effective at distances greater than a mile or two. Deposition will be in the downwind direction and often sufficiently localized laterally so that evacuation crosswind would be successful.

The possibility of effective decontamination, at least in built-up areas, should be mentioned. Fire hoses, with or without detergent, could accomplish a great deal in reducing contamination levels. In lawns or open country, rain would gradually wash activity down into the soil so that the external exposures would be gradually decreased. This process could be speeded by plowing. The activity would remain in the soil and be available for plant uptake until it finally decayed.

In a study of this sort it is obviously necessary to make many arbitrary assumptions and to set up definite levels for evacuation, etc. It is clear that this results in some rather embarrassing inconsistencies. For example, a person in a region contaminated to range II would be evacuated and not permitted to return for over a year at least. A neighbor a few hundred yards away falling in range III might not be evacuated but would receive a very sizeable radiation dose, particularly if his house were of light construction. The person who was evacuated might happen to live in a substantial house and spend most of his time indoors so that he would have received a smaller dose than his neighbour if he had remained.

In estimating the dollar costs of a postulated reactor accident it must be borne in mind that the contamination levels proposed above are not "standards" to be rigidly adhered to. Rather they are guesses as to what action might be taken by public health authorities in the circumstances. This is particularly true of the limits for ranges III and IV. If only a small area were affected, it is likely that the land would be taken out of production. On the other hand, if hundreds of square miles were in question, it would become worth while to try to find out just what crops might safely be grown. Obviously assignment of fixed dollar costs on a per person or per acre basis to the various contamination ranges can only be approximate.

It must be pointed out that several of the limits suggested, particularly those for ranges III, IV and V, are quite sensitive to the strontium-90 levels, and therefore the situation would be much worse for a reactor incident

involving fuel that had been irradiated longer than 180 days as postulated here.

Finally, it should be pointed out again that the attempted estimation of the effects on people and land use of so many different isotopes is only a rough guess. Even with a great deal more data than is now available it would still be very hard to find out how the effects add up.

APPENDIX I

Summary of methods and conclusions for BNL study of public hazards of castrophic accidents in large nuclear power plants

This study was undertaken because of a desire to place a dollar value on the public liability possibly associated with a catastrophic reactor accident. There was considerable previous work but only part of it is generally available.[2–12] The study was based on a hypothetical reactor of 500 thermal MW, equipped with containment shell, that had operated 180 days and whose accumulation of fission products would amount to 4×10^8 c as of 24 hr after the supposed accident. The fission product inventory is shown in Table 2. In accordance with the observed trend for presently proposed plants, the reactor was assumed to be on a watercourse, flanked with low hills, at a distance along the watercourse of 30 miles from a major city. Population density in people per square mile was assumed to increase from the reactor to the city according to the equation $P = 200\ R^{2.83}$ where R is in miles. Beyond the city a population density of 500 people/mile^2 was used. These assumptions are typical of heavily populated parts of the country where nuclear power stations are most likely to be located.

The study group was unable to assess a zero probability for catastrophic accidents but found it difficult to set numerical values for such probabilities. Table 5 shows estimates furnished by those individuals who were willing to express opinions.

Using the criteria on exposure and fallout presented above and the assumptions just summarized, the numbers of people killed, injured or subject to evacuation were determined for a total of thirty-three cases. Three basic types of accidents were considered.

Table 5. Probability estimates in cases per reactor year

Release within the reactor shield	10^{-2}–10^{-4}
Release to the containment shield	10^{-3}–10^{-4}
External release—average conditions	10^{-5}–10^{-7}
External release—extremely unfavorable conditions	10^{-6}–10^{-9}

(1) The contained case where no fission products escaped from the containment shell and off-site damage was due only to gamma exposure.

(2) The volatile release case where all volatile fission product gases and 1 per cent of the strontium were assumed to have escaped from the containment shell.

(3) The major release case where 50 per cent of all fission products were assumed to have escaped from the containment shell.

For each of the latter two cases all possible combinations of the following assumptions were separately utilized.

(1) Hot incident resulting in a temperature of 300°F which is the steam temperature deemed sufficient to breach the containment shell.

(2) Cold incident with a temperature of 70°, i.e. equal to ambient.

(3) Particles typical of fumes with a mass median diameter of 1 μ.

(4) Particles typical of dusts with mass median diameter of 7 μ.

(5) Normal daytime meteorological conditions with a positive lapse rate and a surface wind of 12 m.p.h.

(6) Normal night-time meteorological conditions with a temperature inversion and a surface wind of 7 m.p.h.

(7) No precipitation.

(8) Rain of 0.02 in/hr over the whole area affected.

Cloud rise was taken to be 0 m for the cold case, 860 m for the hot case during lapse conditions and 400 m during inversion conditions. The assumptions and procedures used in computing the diffusion and deposition of fission products have been discussed in a recent article by SMITH and SINGER.[13]

In computing costs, a figure of $5000 was assigned to permanent evacuation of an

individual and $750 to temporary evacuation or restriction of the mode of living. $25,000/mile^2 was charged for crops destroyed. Results for the various cases varied widely of course. Fatalities ranged from 0 to 3400, people injured from 0 to 43,000 and costs from half a million to seven billion dollars. The area involved in restrictions varied from 18 to 150,000 miles^2. Costs for most of the cases do not exceed a few hundred million dollars. Injuries depend strongly on meteorological conditions, being few in the daytime and more numerous at night due to the temperature inversion.

If the highest probability figure quoted above for an external release is combined with the estimated fatalities for the worst case, and the existence of a hundred power reactors in the U.S. is postulated, the death rate due to reactor accidents would be about one per year per fifty-million people. This is to be compared with deaths due to auto accidents which average one per year per five-thousand people. Thus the risk appears to be very small on the average. Locally, hazards will be considerably higher and the costs of a catastrophe, if and when it occurs, may be very large.

REFERENCES

1. *Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants.* United States Atomic Energy Commission (1957).
2. W. G. Marley and T. M. Fry, *Geneva Conference on Peaceful Uses of Atomic Energy.* Paper 8/P/394 (1955).
3. H. M. Parker and J. W. Healy, *Geneva Conference on Peaceful Uses of Atomic Energy.* Paper 8/P/482 (1955).
4. W. F. Libby, *Proc. Nat. Acad. Sci.* **42,** No. 6, 365 (1956).
5. J. Z. Holland, *Geneva Conference on Peaceful Applications of Atomic Energy.* Paper 8/P/572 (1955).
6. C. R. McCullough, M. M. Mills and E. Teller, *Geneva Conference on Applications of Atomic Energy.* Paper 8/P/853 (1955).
7. A. G. Hoard, M. Eisenbud and J. H. Harley, *Annoted Bibliography on Fall-out Resulting from Nuclear Explosions.* Report NYO-4753, NYOO-AEC (1956).
8. G. E. Harrison, W. H. A. Raymond and H. C. Tretheway, *Harwell Report* AERE/SPAR-2 (1955).
9. R. S. Russell, H. M. Squire and R. P. Martin, *Harwell Report* AERE-SPAR-3 (1955).
10. *The Hazards to Man of Nuclear and Allied Radiations.* H.M. Stationery Office, London (1956).
11. A. J. Breslin and L. R. Solon, *Fallout Countermeasures for AEC Facilities: A Preliminary Report.* NYO-4682A, NYOO-AEC (1955).
12. *Pathologic Effects of Atomic Radiation.* Publication 452, National Academy of Sciences, National Research Council (1956).
13. M. E. Smith and I. A. Singer, *Industrial Hygiene Assoc. Quarterly* **18,** 319 (1957).

Health Physics Pergamon Press 1958. Vol. 1. pp. 85–89

METABOLISM OF CESIUM-137 IN THE RAT: COMPARISON OF ACUTE AND CHRONIC ADMINISTRATION EXPERIMENTS*

JOHN E. BALLOU and ROY C. THOMPSON
Biology Operation, Hanford Laboratories
General Electric Company, Richland, Washington

(*Received* 20 *September* 1957)

Abstract—Data were obtained for the retention of Cs^{137} in the rat over a period of 200 days following a single administration; and data showing the buildup of Cs^{137} during a period of 300 days of chronic feeding. Predictions of the chronic buildup results, based on the single administration data, were in close agreement with the results of the chronic feeding experiment. The highest equilibrium Cs^{137} concentration (10 per cent of the average daily intake per gram) was observed in muscle. Retention of cesium in muscle was closely approximated by a single exponential function with a biological half-life of 13 days. There was no evidence, in any tissue, for retention of Cs^{137} with a half-life longer than 17 days.

From the standpoint of radiation hazard evaluation and control the metabolism of Cs^{137} has been comparitively well worked out in a variety of animal species.[1–5] In undertaking further experiments with rats we were therefore interested not so much in a general study of cesium metabolism as we were in seeking the answers to certain specific questions. We wanted to know whether some small fraction of deposited cesium might be retained with a much longer half-life than indicated in earlier studies which were of insufficient length to rule out the presence of a small, long-lived component. We were interested in a closer look at the relative criticality of various organs. We were also interested in conducting long-term chronic feeding studies from which the maximum permissible concentration (MPC) of Cs^{137} in water could be directly inferred without assumptions regarding half-lives, fractions deposited, etc.; and to then compare these MPCs with those calculated on the basis of retention studies following a single administration of the isotope.

* This paper is based on work performed for the Atomic Energy Commission under contract W-31-109-Eng-52.

EXPERIMENTAL

In the single administration experiment thirty-five female Sprague–Dawley rats weighing approximately 240 g were each given, intraperitoneally, 90 μc of Cs^{137}. Animals were sacrificed in groups of three at 1, 3, 7, 15, 30, 60, 100, 150, 200 and 300 days following injection. Tissues were digested in concentrated HNO_3 to give a homogeneous solution and an aliquot was plated for counting with an end window GM tube. Appropriate corrections were made for self absorption where necessary.

In the chronic feeding experiment Cs^{137} was present in the drinking water at a level of 0.002 μc/ml. Water consumption, and therefore Cs^{137} intake, was determined for each animal. Female rats, weighing approximately 240 g at the start of the experiment, were sacrificed in groups of five after 30, 60, 100, 150, 200 and 300 days on the Cs^{137} regimen. Male rats weighing approximately 350 g at the start of the experiment were sacrificed in groups of five after 30, 100 and 200 days. Tissues were analyzed for Cs^{137} as described above.

RESULTS AND DISCUSSION

From the single administration experiment data on retention of Cs^{137} were obtained for eleven organs and tissues over a period of 200 days (Table 1). Samples taken at 300 days contained too little Cs^{137} for accurate assay.

Table 1. Cesium-137 distribution in the rat as a function of time after intraperitoneal administration

Time of sacrifice (days)	1	3	7	15	30	60	100	150	200
Average wt. at sacrifice (g)	243	244	229	252	265	271	284	288	308
	% of administered dose per gram tissue* ($\times 100$)								
Blood	7.6	6.3	2.8	1.4	0.64	0.11	0.014	0.0029	0.0014
Liver	58	29	13	8.9	3.5	0.59	0.091	0.010	0.0023
Kidney	75	35	18	8.6	5.3	0.61	0.066	0.015	0.0018
Heart	57	28	16	8.1	4.8	0.66	0.074	0.011	0.0018
Brain	8.3	11	9.2	3.3	2.1	0.42	0.012	0.0071	0.0015
Lung	39	18	10	5.0	2.3	0.43	0.049	0.0061	0.00068
Muscle	44	50	35	25	11	2.2	0.28	0.035	0.0033
Femur	13	7.2	5.4	2.9	1.5	0.19	0.052	0.12	0.0010
Ovaries	42	24	17	5.7	4.8	0.55	0.050	0.029	0.0021
Spleen	59	22	13	5.3	3.4	0.62	0.069	0.0093	0.0025
Total animal	44	39	25	18	7.1	1.1	0.21	0.019	0.0031

* Each figure is the average of analyses on 3 animals. Data were normalized to a constant rat weight.

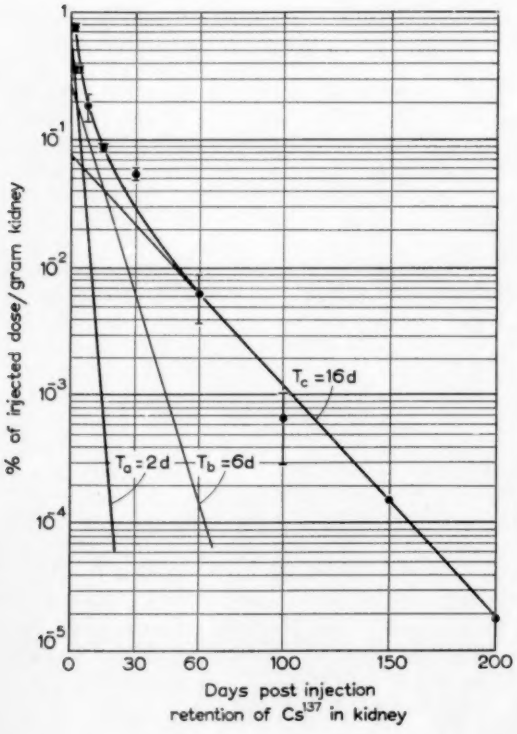


Fig. 1 shows the retention curve for kidney, which was typical of most of the curves obtained. Over a concentration range of nearly 10^5 there is no evidence for any component with a half-life of longer than 16 days. The curve can be resolved quite nicely into three exponential components of 2, 6 and 16 day half-life.

All tissues studied exhibited a component of 15 to 17 day half-life which constituted from 9 to 45 per cent of the total tissue Cs^{137}. A component of 5 to 8 day half-life was present in all tissues and constituted from 19 to 70 per cent of the total Cs^{137}. A component of 1.5 to 2 day half-life was observed for all tissues except muscle and brain and constituted the majority of Cs^{137} present in such organ tissues as liver, kidney, heart, lung and spleen. These half-lives and the magnitudes of the components bearing the half-lives are summarized in Table 2.

Fig. 2 shows the retention curve for Cs^{137}

FIG. 1. Retention of Cs^{137} in kidney (points are indicated with standard deviation. Where no standard deviation is shown pooled samples were analyzed).

Table 2. Cesium-137 retention in rat tissues

Tissue	Retention equation* $[Q = Q_0\{A \exp(-0.69t/T_a) + B \exp(-0.69t/T_b) + C \exp(-0.69t/T_c)\}]$						
	Q_0	A	T_a	B	T_b	C	T_c
Blood	0.077	0.23	1.5	0.60	5.5	0.17	17
Liver	0.63	0.69	2	0.19	7	0.12	16
Kidney	0.79	0.58	2	0.33	6	0.09	16
Heart	0.66	0.53	1.5	0.34	5	0.13	16
Brain	0.15	0	—	0.60	5	0.40	15
Lung	0.41	0.58	1.5	0.27	6	0.15	15
Muscle	0.59	0	—	0.55	8	0.45	16
Femur	0.15	0.27	2.0	0.53	6	0.20	17
Ovary	0.50	0.44	1.5	0.44	7	0.12	17
Spleen	0.63	0.77	1.5	0.14	8	0.09	16
Total animal	0.46	0	—	0.70	7	0.30	16

* Symbols are defined as follows:

Q = concentration of Cs^{137} in tissue at time t (in days) following single administration (units of per cent of administered dose per gram)

Q_0 = concentration of Cs^{137} in tissue at zero time (extrapolated from retention curve) (units of per cent of administered dose per gram)

A, B, C = fraction of Q_0, present at zero time, in components of half-life T_a, T_b, T_c (in days)

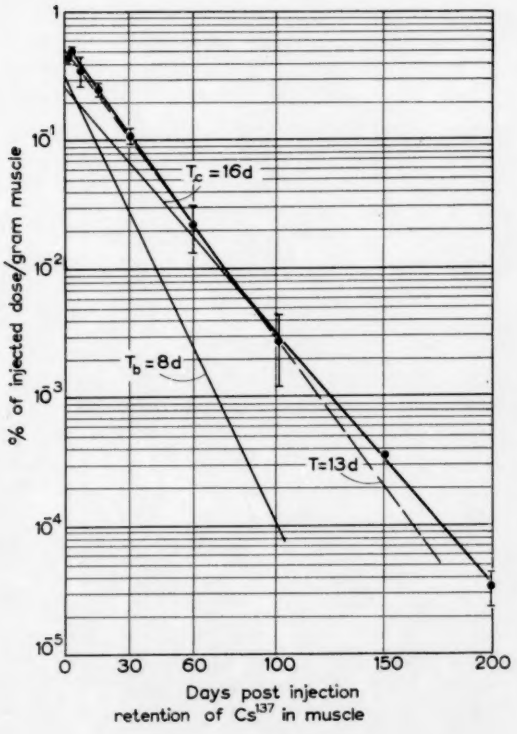


FIG. 2. Retention of Cs^{137} in muscle (points are indicated with standard deviation. Where no standard deviation is shown pooled samples were analyzed).

in muscle. The curve is shown resolved into exponential components of 8 and 16 day half-life. The broken line represents a single component of 13 day half-life which for the purpose of MPC calculation would very adequately represent the retention of Cs^{137} in muscle. Corresponding to the 13 day half-life, the fraction of Cs^{137} which is deposited in muscle amounts to 57 per cent of the total injected dose. Using these parameters and assuming 100 per cent absorption of ingested cesium, the MPC for Cs^{137} in water calculates to 1.8×10^{-3} μc/ml, and is in agreement with the presently recommended value of 2×10^{-3} μc/ml.[6] The presently recommended value is based on a somewhat longer half-life of 17 days, and a smaller fraction deposited of 0.45.

The data in Table 3 indicate that muscle is properly considered the "formal" critical organ. This table compares the equilibrium concentrations of Cs^{137} to be expected in the

Table 3. Buildup of cesium-137 in rats during period of chronic feeding
(Per cent of daily dose/gram tissue)

Tissue	Days on chronic feeding regimen						Level predicted from single dose experiment
	30	60	100	150	200	300	
Blood	0.8	1.0	1.2	1.1	1.0	0.8	0.7
Liver	3.9	4.6	5.0	4.8	5.2	5.4	4.2
Kidney	5.9	7.3	7.6	6.9	6.4	6.4	5.2
Heart	4.5	4.2	5.0	5.1	4.5	4.6	4.4
Brain	1.9	2.0	2.7	2.5	2.4	2.3	1.9
Lung	2.9	2.8	2.8	2.8	2.9	2.9	2.8
Muscle	9.9	7.8	8.5	9.8	10.1	9.4	9.9
Femur	1.3	0.8	1.4	1.6	1.4	1.5	1.5
Ovary	2.2	3.4	2.6	3.3	3.4	3.1	4.2
Spleen	3.2	2.5	3.5	3.6	4.0	3.7	3.4
Total animal	4.1	5.6	6.4	7.2	6.5	7.6	6.4

various tissues, as calculated from the results of the single dose experiment, with the actual Cs^{137} buildup observed in the female rats of the chronic feeding experiment. While muscle clearly shows the highest Cs^{137} concentration, it is interesting to note the relatively high Cs^{137} concentrations in several other tissues. These might well be more critical than muscle from a radiation damage standpoint. There is, in the case of all tissues, quite close agreement between the predicted and observed equilibrium values. Since the muscle results are in almost exact agreement, the MPC calculated from the chronic feeding equilibrium level will be the same as that calculated from the results of the single administration experiment.

One defect in the chronic feeding experiment should be pointed out. While weights of the female rats changes only slightly during the course of the experiment, the drinking water intake, and thus the Cs^{137} dosage, increased quite appreciably as the animals grew older. To eliminate this variable, the "daily dose" employed in the calculation of the results shown in Table 3 was the average daily intake during the 30 days preceding sacrifice rather than the average daily intake during the whole experiment. This procedure seemed justified in view of the demonstrated absence (based on results of the single feeding experiment) of any long-lived cesium retention component.

Results from the male rats exposed chronically to Cs^{137} are not presented in detail, since data were obtained at only three sacrifice periods, and because the substantial weight gains registered during the experiment complicate the interpretation of the results. In general the male data were consistent with the results obtained from the females. Muscle was clearly the critical organ. The content of Cs^{137} in the testes was slightly lower than that in the ovaries of the female rats.

At the 150 and 200 day female sacrifices, the Cs^{137} content of the gastrointestinal tract, and of its contents, was determined. The tissues of the intestine and stomach walls contained a concentration of Cs^{137} approximately equivalent to that of the kidney. The Cs^{137} concentration in the gastrointestinal contents was lower by a factor of approximately five. There is therefore no unusual hazard to the gastrointestinal tract resulting from ingestion of Cs^{137}.

It would appear from the foregoing that the deposition and retention of Cs^{137} is, for hazard evaluation purposes, quite completely

understood, and that the presently recommended MPC is in accord with the experimental data. It must be remembered, however, that this is an MPC based on rat data. Recent studies with humans indicate that these rat data are not directly extrapolatable to man. Using total body counting techniques, RICHMOND and LANGHAM have measured an average biological half-life for cesium in man of 110 days.[5] These human data must clearly take precedence over the half-life determined in the rat. One must still depend on the rat data, however, for information on the distribution of cesium among organs and tissues. The extrapolation from rat to man of information on absorption, distribution, and the general nature of the retention process can probably be made with more certainty than the extrapolation of a specific rate of turnover. If one accepts the general picture of cesium metabolism derived from the rat data, but substitutes the longer half-life measured in humans, then the MPCs would require reduction from presently recommended values by a factor of approximately ten.

It would be well to conclude with a word of caution on generalizing from these results on Cs^{137}. While the results of the present chronic feeding studies with Cs^{137} added essentially nothing to the picture predicted from single administration experiments, and while similar agreement between single and chronic administration experiments has been reported for Co^{60} by COOK *et al.*,[7] it does not follow that such agreement will necessarily exist for other radioisotopes. There is, in fact, reason to believe that these two elements are rather exceptional in the simplicity of their behavior.

REFERENCES

1. J. G. HAMILTON, *Radiology* **49,** 325 (1947).
2. S. L. HOOD, and C. L. COMAR *Arch. Biochem. Biophys.* **45,** 423 (1953).
3. S. A. THREEFOOT, G. E. BURCH and C. T. RAY, *J. Lab. Clin. Med.* **45,** 313 (1955).
4. F. R. MRAZ, M. LENOIR, J. J. PINAJIAN and H. PATRICK, *Arch. Biochem. Biophys.* **66,** 177 (1957).
5. C. R. RICHMOND and W. H. LANGHAM, *Measurement of retention and excretion of radioisotopes of the alkali metals, in mice, rats, monkeys and man.* Presented at annual meeting of Health Physics Society (1957).
6. *Report of international subcommittee II on permissible dose for internal radiation,* International Commission on Radiological Protection, *Brit. J. Radiol.*, Suppl. **6,** 23 (1955).
7. M. J. COOK, K. Z. MORGAN and A. G. BARKOW, *Amer. J. Roentgenol.* **75,** 1177 (1956).

Health Physics Pergamon Press 1958. Vol. 1. pp. 90–91

NOTES

Integrated air sampling program for a radioactive plant

(*Received* 7 *November* 1957)

THE chief hazard in a plant handling radioactive material arises not from external radiation but from inhaled radioactive particulates. This generality does not extend to sources of radiation such as nuclear reactors, particle accelerators and telemetering devices, nor to radioactive gases such as tritium or argon. However, in most industrial plants, it is the airborne contamination levels that must be controlled to reduce the personnel hazard to prescribed values. When the Rocky Flats Plant was constructed several years ago, an air sampling system was installed which has proved quite satisfactory. This system consists of an extensive network of sampling points within each radioactive operating area, a number of outdoor sampling points on the plant site, and several sampling stations in inhabited areas in the vicinity of the plant. The same equipment for drawing a controlled volume of air through a filter paper and similar procedures for counting the sample are used in each area.

In the design and construction of the plant, a sampling station was prescribed for each pertinent area or on 20 ft centers, whichever was consistent with the status of the engineering design at the time. An attempt was made to place at least one sampling station in each room and to cover each machine or other specific location where particulate matter might be generated. An attempt was also made to place the sampling point approximately in a position representing the breathing zone without introducing an objectionable protrusion into the working area. Each sampling point was connected through a system of branch lines and manifolds to a 250 ft^3/min Hoffman centrifugal pump. Each of these pumps appears to be quite capable of sustaining 100 sampling points without difficulty at 2 ft^3/min.

This permanently installed system is supplemented by a number of portable units, Fig. 1, which use the same type sampling head. These portable units are powered by a Gast type unit pump, Model Number 04565-V2-25.

In erecting stations off the plant site, it was considered discreet to make them as inconspicuous as possible and not to approach landowners for permission to install them. Accordingly, arrangements were made with local electric power companies to install them on power poles. The meter is furnished by the utility company and charges are made on the basis of ordinary domestic service (Fig. 2 shows such an installation). The meter is at a level where it can be read easily. The pump in its shelter is placed sufficiently high that it is inaccessible to the inquisitive but can be easily serviced from a short stepladder. An elapsed time meter in series with each pump allows the calculation of the volume of air sampled in case of power or pump failure.

Before the pumps are put into service, their vacuum is checked with either a manometer or Magnehelic gauge. When in operation the pumps pull 2 ft^3/min, or approximately 120 lineal ft/min at the face of the filter paper. Unit failures in operation in several cases were attributed to the coupling mechanism between the pump and motor and for this reason a change in pumps has been made recently to an integral unit (motor-pump) Gast Model 0320-V3-10X $\frac{1}{4}$ h.p., 2.5 ft^3/min to 28 in. vacuum.

The sampling stations in buildings are serviced daily to yield a continuous record of the working conditions. Those on the plant site are serviced weekly in order to discover if there has been any undetected escape of radioactive material from the ventilation filter banks. The more remote sampling stations are serviced fortnightly.

The sampling head is a plastic cylinder closed by a bayonet lock on a rubber "O" ring seal. These sampling heads were manufactured on order by the Chaney Plastics Moulding Company, 4200 Madison, Denver, Colorado. The off-site sample heads are equipped with 5.5 cm Whatman No. 41 filter paper which is harder than the paper used in the other sampling heads and capable of maintaining a greater dust load without ripping. The filter paper used in the plant site and building sample heads is 18 mil Hollingsworth and Vose No. 70. The air sample head is designed to take a $2\frac{1}{8}$ in. diameter disc which fits the hemispherical chamber of the Nuclear Measurements Corporation PC-2 gas flow proportional counter, or the slide of the Eberline Instrument Division multi-channel counter.

The specifications of the gas flow proportional counter state that it is not suitable for the alpha

FIG. 1. Portable sampling device.

FIG. 2. A remote sampling station.

analysis of paper-borne samples due to the fact that the paper acts as an insulator between the ground of the counting chamber and the sample. Also, because of this self-adsorption of the sample and the characteristics of the particle range, a well-defined plateau does not exist. Accordingly, a number of gas flow proportional counters have been converted into scintillation counters[1] by substituting a 5819 photomultiplier tube, preamplifier, Lucite light pipe and silver-activated zinc sulfide phosphor for the gas flow mechanism. The Eberline Instrument Division has designed and built a multi-channel counter to The Dow Chemical Company specifications incorporating this feature.

In all samples, there exists a background of dust-borne radioactivity which arises from the decay of isotopes of the inert gases, principally radon and thoron. The radon chain is controlled by the 26.8 min radium β and the thoron chain is controlled by 10.6 hr thorium β. Because of the activity arising from the decay of these isotopes and their daughters, it is customary to wait at least 24 hr before counting the filter disc. This permits the complete disappearance of the radium daughter activity and the decay of more than three-quarters of the thorium daughter activity. This delay is usually sufficient to render the background negligible but occasionally it has been found necessary to recount after a longer elapsed time in order to assure that the activity recorded is entirely due to the thorium daughter background.

In order to study personnel exposure incident to specific operations, it is necessary to measure the particulate activity in the breathing zone of the operator. For this purpose, a standard sample head is mounted on a Bullard head band such as is used to support the hood for a supplied air mask. The air line from this sampling head may be connected with the house vacuum outlet which is usually convenient or, for mobility, to a portable sampling unit.

It is usually advantageous to supplement the system with a high-volume air sampler. Such a supplementary unit makes it possible to obtain a more rapid determination of airborne activity. A rapid determination of the working conditions for a novel operation or for contamination following an incident may be obtained by operating the high-volume sampler for a short period and scanning the face with a portable survey instrument.

The Dow Chemical Company, T. S. CHAPMAN
Rocky Flats Plant,
Operated by USAEC
Contract AT/29-1/-1106

REFERENCE

1. J. R. MANN, *Health Physics* **1,** In press (1958).

Announcement of the Annual Meeting of the Health Physics Society

THE annual meeting of the Health Physics Society will be held at the University of California, Berkeley, California on 9, 10 and 11 June. The meeting will be similar to those of the last two years with technical sessions from Monday morning until Wednesday noon and will be open to everyone interested in the field of health physics upon payment of a nominal registration fee.

Papers from the meeting will be eligible for publication in *Health Physics* if they meet the journal's standards.

Proceedings of the Health Physics Society

THE published proceedings of the First Annual Meeting of the Health Physics Society, held at the University of Michigan in June 1956, are still available. This volume contains thirty-four complete papers and abstracts of six papers that cover a wide range of health physics activities. It may be obtained for $3.00 per copy from Radiation Control Service, School of Public Health, University of Michigan, Ann Arbor, Michigan.

LETTERS TO THE EDITOR

Suggested insignias for the Health Physics Society

(*Received* 28 *November* 1957)

DURING the course of the Pittsburgh meeting of the Health Physics Society, I noted that we were considering possible symbols for the Society which could be used on stationery, insignia, membership cards, the Journal, etc. Based upon my observations then, and some previous experience in attempting to develop symbolic forms for the Forum, I attempted to put a suggestion on paper.

Enclosed please find a representation of my suggestion for an insignia for the Society. The symbolism intended to be conveyed by this drawing is based upon the basic approaches to radiation control; time, distance and shielding. Of these, time plays the most important part. Hence, the basic design is that of a clock face. Time also plays a part in the application of permissible limits and is also indicative of measurement.

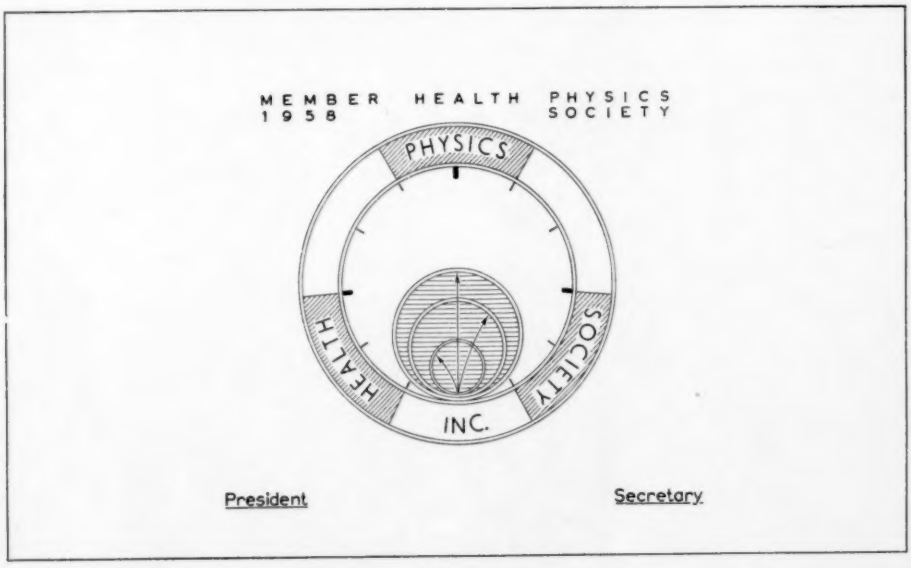
The three "arrows", where the seconds hand would be on a clock, are symbolic of alpha, beta and gamma radiations in the paths they follow upon passing through a magnetic field. These paths are widely used to represent radiation. The three circles around each "arrow" cut each off in succession, showing alpha as least penetrating, beta as medium penetration and gamma as the most penetrating. It is suggested that if this design is to be considered further, the area of the circles should also vary symbolizing the inverse square law. A more competent artist might also draw the widths of the circumferences of each circle of varying thickness to designate shielding requirements. The neutron is not shown on this symbol, but may be worked in some other way.

Behind the clock face, and extending to the outermost circle, might be a bare suggestion of the radiation hazard symbol. The name of the Society would be printed in the general location as indicated, upon the segments of the hazard symbol which come through. At the bottom would be the designation "Inc." or the date the Society was first formed.

I would welcome comments on this proposed design and would also like to suggest that all members of the Health Physics Society send in any proposals for the Society's symbol as soon as possible. We all know how important it is for an organization to have for general use a symbolic representation of its purposes. I am therefore offering this rendition as a start along these lines.

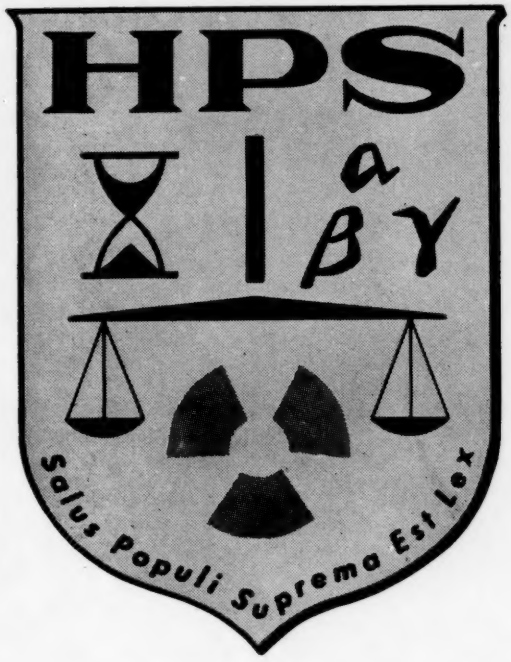
SAUL J. HARRIS

Assistant Manager of
Technical Services
Atomic Industrial Forum, Inc.



(*Received* 26 *December* 1957)

I AM submitting herewith a proposed symbol for the Health Physics Society.



Consideration for use of this symbol, review, suggestions or comments would be appreciated.

Head, Health Physics Division A. L. BAIETTI
U.S. Naval Radiological Defense Laboratory
San Francisco 24, *California*

THE following emblem suggestion was submitted to the Health Physics Society secretary in October 1956, by S. G. FIDLER, who is now with Wright Air Development Command. The snake was used to depict: "first, the deadliness of nuclear radiations and secondly, the health or medical tie-in as depicted on military medical insignia".

Health Physics Pergamon Press 1958. Vol. 1. p. 94–95

NEWS

Health Physics in the News

THE following news items reporting on significant developments in Health Physics have been selected by Mr. S. J. HARRIS from newspapers, press releases and technical journals issued during the last quarter.

Public fears of radiation

In December the World Health Organization announced a plan suggested by a special study group "for community education in atomic energy questions, to guard against irrational fears and irrational hopes." The study group reported that it had found that the first uses of atomic energy "had aroused a deep sense of fear" against a possible "biological chain reaction" from increasing uses of radiation.

State control of radiation

The Joint Federal–State Action Committee on 5 December submitted a report to the White House which recommended, among several other items, the giving of greater authority to States in the areas of public health and safety, inspection, and in the enforcement of radiation standards. The Committee recommended that the President urge Congress to amend the Atomic Energy Act of 1954 to carry this out. The report also urged that State governments establish standards for radiation control.

AEC revises limits for contractors

On 10 December, the AEC set the permissible limits of accumulated dose for atomic workers employed by the Commission and its contractors and to populations outside Commission owned facilities at one-third the integrated maximum permissible dose rate. This action does not affect AEC licensees, although the Commission announced it "is currently considering amendments to this regulation (Title 10, Chapter 1, CFR, Part 20) to make it consistent with the new recommendations." The AEC also stated, in its announcement, that it was approving the recommendations made by the National Committee on Radiation Protection and Measurement in taking this action. The new limit for occupational exposure is essentially the 5 rem per year average from age 18 with no more than 15 rem in any one year.

Pills for radiation?

Last quarter, Dr. W. F. LIBBY announced that the AEC was investigating the use of calcium pills to decrease the retention of strontium-90 in bone. The research was reported in the 6 December 1957 issue of *Science*.

Windscale Incident

On 23 November, forty-four days after the onset of the Windscale Incident, normal collection and distribution of milk in the banned (200 square miles) area was resumed. The Prime Minister's report to the British Parliament indicated that no person on-site or off-site received exposures greater than emergency maximum permissible exposures.

Castastrophe Committee

As a result of the Windscale Incident, a Catastrophe Committee is being formed by the AEC to strengthen emergency plans such that incidents of the Windscale type will be avoided or the results of such will be significantly lessened.

ASA activities on Nuclear Standards

The American Standards Association N–2 Committee on General and Administrative Standards reported progress on organizing sub-committees on Radiation Exposure Records Systems and Qualification of Nuclear Professionals. The committee postponed formal action on a uniform radiation symbol because several transportation groups and the International Labour Organization strongly favor other symbols.

The ASA N–7 Committee on Radiation Protection has organized committees on radiation protection in mines and mills, safety standards in refineries, isotopic separations protection, radiation protection for fuel element fabrication and radiation safety in reactor plants.

International activities on Health Physics Standards

The General Secretary of International Standardization Organization (ISO) has offered the services of its Technical Committee–85 to the International Atomic Energy Agency. The IAEA has a responsibility to set international standards for safety and

health in the design and location of reactors and the disposal of radioactive wastes.

The British Government has appointed three committees to determine what improvements are needed in the atomic energy facilities of the country to prevent future incidents of the Windscale type. Sir ALEXANDER FLEAK, Chairman of Imperial Chemical Industries, Ltd, has been named Chairman of the project. The three committees were established at the suggestion of the U.K. Atomic Energy Authority and will deal with technical evaluation organization and health and safety.

The World Health Organization is developing a new community education plan to allay unwarranted fears about the hazards of atomic energy projects. The plan involves an organization which includes a psychiatrist, psychologist, sociologist and a journalist to aid in the persuasion of the public to accept new nuclear projects.

State activities in Health Physics

State health officers at their annual meeting with the Surgeon General at the Department of Health, Education and Welfare were urged to develop an effective radiation protection program in their respective states. The Surgeon General, LEROY E. BURNEY stated that Public Health has not yet taken its proper place in radiation protection.

THE SECOND INTERNATIONAL COURSE IN HEALTH PHYSICS

THE Second International Course in Health Physics was held at the Centre D'Etudes Pour Les Applications de L'Energie Nucleaire, Mol, Belgium, from 30 September to 31 October 1957. The First International Course was held at the Radiophysics Institute, Stockholm, Sweden, from 14 November to 16 December 1955. The Course was sponsored by the European Office of the World Health Organization, the Belgian Government, and the United States Atomic Energy Commission, and was under the direction of Dr. ELDA E. ANDERSON. The course was conducted by E. E. ANDERSON, M. F. FAIR and L. C. EMERSON—all members of the Health Physics Division, Oak Ridge National Laboratory. The material presented in the Course included biological effects of ionizing radiation, dose units, instrumentation, external and internal exposure, radiation protection, radioactive waste disposal, basic reactor theory and source and fissionable material processing. Many specific problems in Health Physics were discussed in informal sessions.

The twenty-six (26) participants were from the following countries: Austria, Belgium, Bulgaria, Egypt, France, Germany (West), Greece, Iran, Iraq, Italy, Lebanon, Netherlands, Poland, Portugal, Russia, Switzerland, Turkey, and Yugoslavia.

Guest lecturers included staff members of the Centre D'Etudes as well as Dr. WALTER CLAUS, Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington, D.C., Dr. B. A. J. LISTER, Atomic Energy Research Establishment, Harwell, England, and Mr. PER GRANDE, Norwegian Radium Hospital, Oslo, Norway. The main address of the closing day ceremonies was given by Professor Z. BACQ, University of Liège. Perhaps the most significant contribution of the course, in addition to increasing knowledge of Health Physics, was the promotion of international goodwill.

PAPERS TO BE PUBLISHED IN FUTURE ISSUES

K. Z. MORGAN: Current status of the internal dose problem.

R. L. KOONTZ and C. T. NELSON: A continuous automatic environmental air sampler.

P. S. HENSHAW: Assessing the impact of environmental radiation on total health.

F. P. COWAN and J. B. H. KUPER: Exposure criteria for evaluating the public consequences of catastrophic accidents in large nuclear power plants.

W. DE LAGUNA: Desert basins and waste disposal.

S. G. PEARSALL, L. GEMMELL and A. BRESLIN: Possible health hazards associated with the laundering of radioactively contaminated protective clothing.

P. R. BOLTON, E. C. BARNES, E. D. DURKOSH and W. D. SMALL: Aerosal activity from experimental low power reactor operation.

W. LANGHAM: Potential hazard of world-wide Sr^{90} fallout from nuclear weapons testing.

K. E. COWSER and F. L. PARKER: Soil disposal of radioactive wastes at ORNL: Criteria and techniques of site selection and monitoring.

J. R. MANN: Converting a gas flow alpha proportional counter to the scintillation type.

J. NEUFELD: On the relationship between the charge of an ion and its velocity.

G. W. C. TAIT and W. F. MERRITT: Emergency radiation monitoring of drinking water.

P. W. REINHARDT and F. J. DAVIS: Improvements in the threshold detector method of fast neutron dosimetry.

L. S. TAYLOR: History of the International Commission on Radiological Units and Measurements (ICRU).

L. S. TAYLOR: History of the International Commission on Radiological Protection (ICRP).

P. J. BARRY: Some general considerations in chemical decontamination.

A. BRODSKY, L. W. FAGG and T. D. HANSCOME: A short procedure for electrodepositing fissionable elements.

Health Physics Pergamon Press 1958. Vol. 1, pp. 97–104

HISTORY OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION (ICRP)

LAURISTON S. TAYLOR*

(*Received* 31 *March* 1958)

Abstract—The ICRP was established by the second International Congress of Radiology in 1928. Since this time the Commission has established the basic pattern for radiation protection recommendations that have gone into use thoughhout the world. The first quantitative value of permissible dose was promulgated in 1934 and remained in effect until 1950. In 1950 the MPD was lowered by a factor of about 2, and in 1956 it was again lowered by a factor of about 3. In addition to the development of the basic concept of MPD, the Commission has also adopted maximum permissible concentrations of radioactive materials in air and water and the whole body. The discussion below outlines the various steps that have taken place in the development of radiation concepts since the establishment of the Commission just 30 years ago.

1. FORMATION

THE ICRP was formed in 1928 under the auspices of the Second International Congress of Radiology then meeting in Stockholm. The formation of an international protection committee had been discussed by the First Congress in 1925 but no active steps had been taken at that time toward this end.

During the Second Congress, initial plans for organizing an international protection committee were developed by G. W. C. KAYE and S. MELVILLE (Great Britain), and L. S. TAYLOR (U.S.). Dr. G. GROSSMAN (Germany) and Dr. R. SIEVERT (Sweden) were brought into the discussions and plans were completed for the organization of a committee. Their proposals and organization plans were presented to the Second International Congress and were approved. The committee was named the International X-ray and Radium Protection Committee, and its membership consisted of the five individuals who planned the organization.

2. MODE OF OPERATION AND HISTORY

Because of the early experience of the ICRU with its large and unwieldy membership, the Committee decided at the outset to keep its membership as small as possible. It also decided to include in its membership only people who were working actively in the general field of radiation protection. For this reason, the initial Committee consisted of only five persons. This was enlarged for the next Congress by adding Dr. I. SOLOMON of France and Dr. CERESOLE of Italy.

The ICRP, because of its small size, was operated much more informally than the ICRU. The chairman from 1928 through 1937 was Dr. G. W. KAYE, then of the National Physical Laboratory, and the honorary chairman Dr. S. MELVILLE. It might be remarked (with some interest in view of later events) that at the time of its formation the problem of radiation protection was not considered to be as important or as serious as that of radiation units. Members of the ICRP felt that the problem would grow and sought to be prepared.

1928

The Committee held its initial meeting in 1928 for the purpose of adopting some interim protection regulations. Until that time, the only clearly formulated recommendations were those prepared a few years earlier by the British Committee.[1] These were used as a basis for discussion. The first recommendations of the

* Member, ICRP.

Committee[2] were very similar to the early British proposals, and until 1950 its recommendations were patterned around these early British recommendations.

1931

The second meeting of the Committee took place in Paris in 1931 and was attended by all seven members. At this meeting, there was some discussion of the possibility of introducing so-called tolerance levels of radiation for radiation workers. This was recognized as desirable, but with very little evidence to go on, no specific recommendations were made.[3]

1934

The third meeting took place in Zurich in 1934 and was again attended by all seven members. This meeting was notable because there was established for the first time a permissible level of radiation exposure. This was related to the newly established unit of X-ray dose—the roentgen. The level then set was 0.2 r/day.[4] (It might be noted that the year following this, the National Committee on Radiation Protection in the United States recommended a lower level of 0.1 r/day.)

1937

The fourth meeting was held in Chicago in 1937 with five of the original seven members attending.

Throughout the period 1928 through 1937, the main work of the Committee consisted of gradually enlarging and extending the recommendations originally developed at its first meeting. Until then, the main item of the protection recommendations was the specification of the thicknesses of barriers of various materials that should be interposed between a source and an individual in order to assure adequate protection. Prior to the establishment of the permissible dose in 1934, these barrier thicknesses had been somewhat arbitrary. In 1934 and again in 1937,[5] the relationship of the barrier thickness to the actual dose in roentgens was clarified and was put on a sound quantitative basis.

Following the deaths of both the chairman and the honorary chairman, L. S. TAYLOR acted as secretary for the Committee until its next meeting which was not to occur until 1950.

Under circumstances similar to those of the ICRU, the ICRP was reorganized in 1950 on a different basis from that which prevailed before the war. By that time, the questions of radiation protection had assumed far greater importance than they had earlier. This, of course, was due to the advent of atomic energy. The Committee was enlarged in size to twelve members in order to better deal with the problems facing it. The new membership was made up of individuals having recognized standing in the field without regard to nationality, and its operating rules were made the same as those of the ICRU. At this time, its name was changed to International Commission on Radiological Protection.

1950

The ICRP held its fifth meeting in London in 1950, attended by nine members and a chairman. Between 1946 and 1950, a large body of protection information had been prepared by the NCRP in the United States, having to do with exposure of the whole body to radiation from both external and internal sources. The NCRP had also developed considerably more details in the general field of X-ray protection. This large background of information was used as a basis for discussions of the ICRP which resulted in the development of its 1950 report. It should be mentioned that an extensive interchange of this information had previously been accomplished through the medium of the Tripartite Conference between the United States, Canada, and England. The results of these conferences were in turn based to a considerable extent on data developed during the Manhattan District days.

With the availability of this new information, the ICRP lowered the maximum permissible dose for radiation workers from 0.2 r/day (in use since 1934) to 0.3 r/week. The report[6] also included for the first time maximum permissible concentrations in the body of approximately ten radioactive isotopes.

Because of the very large task confronting it, the IRCP established subcommittees to deal with special phases of the radiation protection

problem. This arrangement has greatly facilitated the development and presentation of information.

The ICRP adopted for the first time a set of rules governing the selection of members and the work of the Commission. These were in agreement with the rules adopted at the same time by the ICRU.[6]

1952

An interim meeting of the ICRP was held in Stockholm in 1952, with the ICRU and the UNESCO Joint Committee on Radiobiology. At this time, the major subject under consideration was the possible genetic effects of radiation. The meeting was attended by a number of geneticists from several countries. At this time, it was recognized that the genetic effects of radiation were becoming better understood and should be taken into consideration. Figures were discussed for an average per capita dose for a large population. The proposals for an average per capita gonadal dose ranged from 3 r (U.K.) to 20 r (U.S.) and a middle value of 10 r was informally accepted. These were in addition to natural background radiation. However, because there was very little agreement between the geneticists themselves, and because of the meager information available on this question, it was decided not to make any specific recommendations. (It is interesting to note, however, that the recent recommendations made on this question in April 1956 by the ICRP and later by the National Academy of Sciences (U.S.) and the Medical Research Council (U.K.) give approximately the same range of figures as those discussed at the Stockholm meeting in 1952.)

1953

The sixth meeting of the ICRP was held in Copenhagen in 1953. This was a joint meeting with the ICRU and was held during the week preceding the opening of the International Congress. Most of the representatives on the subcommittees also met.

The report of the Commission[7] as developed during these meetings marked a radical departure from all others. Its total size was 92 pp. and it included a great amount of detail never before agreed upon on an international basis.

The main body of the report consisted essentially of the reports of its five subcommittees:

(1) Permissible Dose for External Radiation.
(2) Permissible Dose for Internal Radiation.
(3) Protection against X-rays Generated at Potentials up to 3 MV.
(4) Protection against X-rays above 3 MV, β-rays, γ-rays, and Heavy Particles including Neutrons and Protons.
(5) Handling and Disposal of Radioactive Isotopes.

The recommendations on permissible dose from external radiation were somewhat broadened over those previously made, but the basic permissible exposure level of 0.3 r/week was reaffirmed.

Subcommittee 2 gave the maximum permissible concentrations of radioactive materials in the body, air, and water for nearly 100 radioactive isotopes, and included a large body of reference material.

Subcommittee 3 produced a very detailed report applying to both medical and industrial radiology, and included recommendations regarding radiation levels from special devices such as television receivers.

Subcommittees 4 and 5 issued brief reports; much of the material in their areas was still in an unsatisfactory state at that time.

With the development of this detailed report, a new problem arose. Previously the reports had been relatively short and could be published as a few pages in any of the technical journals. However, this report was too large to be published in this manner, and for some time there was uncertainty as to whether or not it could be published in its entirety. Eventually it was published as a supplement to the *British Journal of Radiology*.[7] This had to be underwritten, and for this purpose, some funds were obtained from the British Electrical Industry.

Further difficulty arose because of the expense of carrying out the duties of the secretariat of the ICRP. This expense was borne first by the National Physical Laboratory and then by the British Medical Research Council. All other expenses of the operation were handled

on a voluntary basis with the parent organization of the individual members bearing their expenses.

1956

The seventh meeting of the ICRP took place in Geneva in March 1956, and was held jointly with the ICRU over a period of 12 days.

The report developed in 1953 was not changed in principle. However, greater emphasis was placed on the earlier cautions to keep radiation exposure levels as low as possible regardless of the weekly limits permitted. It was specifically recommended[8,9] that the accumulated exposure of occupational workers be kept below 50 rems up to age 30, and that for the same period the average per capita dose for the population be limited to about 10 rems in addition to background. The compilation of permissible concentrations of radioactive isotopes in the body and in air and water was considerably enlarged to include nearly 200 radioactive isotopes. A new subcommittee was established, in place of the former Subcommittee 5, to deal with the problems of radioactive waste disposal.

During these meetings, the Commission's affiliation with the World Health Organization as a "non-governmental participating organization" in the WHO program was ratified. This affiliation came about as a result of contacts made in the fall of 1955 between the World Health Organization and the ICRP with a view to determining if the latter would participate in the WHO activities. The matter was taken up with the members by correspondence, and it was agreed that the ICRP should give its assistance to WHO. Under the arrangement made, the WHO will not influence the operation of the ICRP but will look to it for guidance in matters of radiation protection.

During the meetings, the relationships between the ICRP and the ICRU were more firmly established. They have always operated as separate commissions but since their inception they have always met at the same time and have maintained close collaboration. There has also been a certain degree of overlap in membership. There is now a clear statement of policy that the two Commissions will work together and consult one another on matters of common interest.

There were also extensive discussions regarding the possibility of forming an International Protection Organization, to be privately financed so as to avoid any nationalistic or political influence. If such an organization were formed, the ICRU and ICRP would be blended into the one organization. No attempt was made to work out the details of such an arrangement, but no difficulties in doing this are foreseen.

It was agreed that if such an organization were established, it should have some semi-official ties with WHO and UNO, without becoming a part of their organization or subject to their direct control. It is felt that one of the strongest points of both Commissions is their freedom to select membership without regard to matters of national politics or other such influences.

Following the meetings of the ICRP in Geneva, a $3\frac{1}{2}$ day joint session was held by some of its members, some members of the ICRU, and representatives from the WHO. The main purpose of this meeting was to outline to WHO the nature of the ICRP work and to point out ways in which WHO might assist in the world-wide implementation of the recommendations of the Commission. A number of specific recommendations were made to WHO and it is believed that they are taking steps to follow these through.

As of the time of writing, the full 1956 report of the Commission has not been officially released.[10] There is again the problem of financing the publication costs. In this connection, WHO has offered to indirectly subsidize the publication by guaranteeing the purchase of 500 copies of the report; this will offset the cost to some extent.

As a result of an inquiry from the United Nations Scientific Committee on the Effects of Atomic Radiation, an emergency joint meeting of the ICRP and ICRU was held in New York from 31 October to 6 November 1956. At this time the Commissions received the following formal resolution from the UNSC and agreed to comply therewith:

(a) To consider and discuss the question of

how to arrive at reliable data indicating the doses to different parts of the body (particularly the gonads) received by individuals and, in the aggregate, by large population groups due to the medical use* of ionizing radiations.

(b) To examine what recording system, if any, is at present feasible for the determination of the relevant dose values; and

(c) As a result of these studies, to submit to the Scientific Committee on the Effects of Atomic Radiation as soon as possible, and in any event before 1 September 1957, a report upon their deliberations and conclusions on the subjects (a) and (b), and to make any appropriate recommendations.

To facilitate the study, the UNSC provided the Commissions with the sum of $8000 for partial defrayment of travel costs and administrative expenses.

To carry out the study a special study group was made up of members from each Commission. Parallel task groups, operating under leadership of the two Commissions, were set up in a number of individual countries. Secretary of the Special Study Group was E. E. SMITH acting in behalf of W. BINKS. Preliminary results of the study were completed by March 1957 and included relatively complete reports from twelve countries.

A meeting of the Study Group was held in Geneva from 23 April to 1 May 1957, at which time the consolidated final report was completed except for editorial details. It was circulated twice to all members of the main commission of the ICRP and ICRU for final approval and was then forwarded to the UNSC before the deadline of 1 September 1957. The report was published within a short time.[11]

As noted earlier, the ICRP and ICRU were established and thereafter sponsored by the International Congresses of Radiology. In a sense the two Commissions were the only continuing bodies of the congresses since the latter existed technically only every 3 years. To some degree, this was awkward for the Commissions in that their sponsorship appeared to be uncertain. This situation was alleviated by the organization of the International Society of Radiology in 1956. This society will in the future, sponsor the International Congresses of Radiology, the ICRP and the ICRU. By its own international character its sponsorship of the Commissions will carry more weight. As a further outgrowth of the new organization it has made funds available to the Commissions to assist in their work. This will be continued in the future and the amounts will probably increase as the Society membership grows.

* For the purpose of this report, and in order to include all types of diagnostic and therapeutic exposure to ionizing radiation, the term "medical use" is extended to apply to all types of medical exposure (except occupational) and the term "radiation practitioner" is used to include radiologists, general practitioners, dentists, obstetricians, osteopaths, chiropractors, etc., who administer medical exposure to ionizing radiation.

REFERENCES

1. *Proposals from the British X-ray and Radium Protection Committee to be submitted for international approval by the II International Congress of Radiology* (Stockholm, 1928). (No publication reference; copy on file).
2. *X-ray and Radium Protection.* NBS Circular 374 (1929); *Brit. J. Radiol.* **1,** 358 (1928).
3. X-ray and Radium Protection, *Radiology* **21,** 212 (1933).
4. International Recommendations for X-ray and Radium Protection, *Radiology* **23,** 682 (1934).
5. International Recommendations for X-ray and Radium Protection, *Radiology* **30,** 511 (1938).
6. *Recommendations of the International Commission on Radiological Protection,* National Bureau of Standards Handbook 47 (1951); *Brit. J. Radiol.* **24,** 46 (1951).
7. Recommendations of the International Commission on Radiological Protection, *Brit. J. Radiol.* Suppl. 6 (1955).
8. Réduction indispensable de la dose de tolérance recommandée par la Commission Internationale de Protection Radiologique, *C. R. Acad. Sci. Paris* **242,** 3134 (1956).
9. Die Internationalen Kommissionen fur Strahlenschutz und fur Radiologische Einheiten und Strahlenmessung, *Dtsch. med. bosch.* **81,** Nr. 26, 1061 (1956).
10. Report on 1956 Amendments to the Recommendations of the ICRP, *Radiology* **70,** 261 (1958).
11. Exposure of man to ionizing radiation arising from medical procedures, *Phys. Med. Biol.* **2,** 107 (1957).
12. *Hearings before the special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the U.S.,* pp. 851–852. (U.S. Government Printing Office (1957).)

ADDENDA

1. *Formal meetings of the ICRP*

Listed below are the formal meetings of the ICRP held since its inception, the number of members in attendance, and the Commission

Date and place of meeting	Members Present	Officers
1928, Stockholm	5	G. W. C. KAYE (U.K.), Chariman
1931, Paris	7	G. W. C. KAYE (U.K.), Chairman
1934, Zurich	7	G. W. C. KAYE (U.K.), Chairman
1937, Chicago	7	G. W. C. KAYE (U.K.), Chairman
1950, London	9	Sir E. ROCK-CARLING (U.K.), Chairman L. S. TAYLOR (U.S.), Secretary
1952, Stockholm	7	Sir E. ROCK-CARLING (U.K.), Chairman L. S. TAYLOR (U.S.), Secretary
1953, Copenhagen	12	Sir E. ROCK-CARLING (U.K.), Chairman W. BINKS (U.K.), Secretary
1956, Geneva	12*	Sir E. ROCK-CARLING (U.K.), Chairman W. BINKS (U.K.), Secretary
1956, New York (with ICRU)	9	R. SIEVERT (Sweden), Chairman G. FAILLA (U.S.), Vice-chairman L. S. TAYLOR (U.S.), Co-chairman E. E. SMITH (U.K.), Acting Secretary

* Main Commission only. Subcommittee membership totalled approximately 50 and a substantial number were in attendance.

Officers. (Officers listed were in office at the time of meeting. Terms of new officers begin after confirmation of their election during the International Congress, and continue for a 3 year period corresponding to the interval between Congresses. Honorary chairman are not listed.)

The Eighth International Congress of Radiology is scheduled to meet in Munich in 1959. The next formal meeting of the ICRP has not been scheduled but will probably be held prior to the Congress. For the period 1956–1959, the officers of the Main Commission are R. SIEVERT (Sweden), Chairman; G. FAILLA (U.S.), Vice-chairman; B. LINDELL, (Sweden) Acting Secretary.

2. *Members preparing reports of the ICRP*

1956–57, New York and Geneva
(Special study for UNSC)

R. M. SIEVERT, Chairman ICRP (Sweden)
G. FAILLA, Vice-chairman ICRP (United States)
E. E. SMITH, Acting Secretary ICRP (Great Britain)
L. S. TAYLOR, Chairman ICRU (United States)
L. H. GRAY, Vice-chairman, ICRU (Great Britain)
H. O. WYCKOFF, Secretary ICRU (United States)
A. ALLISY (France)
W. G. MARLEY (Great Britain)
K. Z. MORGAN (United States)
E. A. WATKINSON (Canada)
(Note: The report was prepared by the above special Study Group but was approved by the Main Commissions of both ICRP and ICRU).

1956, Geneva

Sir E. ROCK-CARLING, Chairman (Great Britain)
W. BINKS, Secretary (Great Britain)
L. BUGNARD (France)
G. FAILLA (United States)
H. HOLTHUSEN (Germany)
J. C. JACOBSEN (Denmark)
R. G. JAEGER (Germany)
W. V. MAYNEORD (Great Britain)
K. Z. MORGAN (United States)
R. M. SIEVERT (Sweden)
R. S. STONE (United States)
L. S. TAYLOR (United States)

1953, Copenhagen

Sir E. ROCK-CARLING, Chairman (Great Britain)
J. C. JACOBSEN, Hon. Chairman (Denmark)
W. BINKS, Secretary (Great Britain)
A. J. CIPRIANI (Canada)
G. FAILLA (United States)
H. HOLTHUSEN (Germany)
R. G. JAEGER (Germany)

W. V. MAYNEORD (Great Britain)
K. Z. MORGAN (United States)
R. M. SIEVERT (Sweden)
R. S. STONE (United States)
L. S. TAYLOR (United States)
M. TUBIANA (France)

1950, London

Sir E. ROCK-CARLING, Chairman (Great Britain)
L. S. TAYLOR, Acting Secretary (United States)
W. BINKS (Great Britain)
E. L. CHERIGIE (France)
A. J. CIPRIANI (Canada)
R. G. JAEGER (Germany)
W. V. MAYNEORD (Great Britain)
R. R. NEWELL (United States)
R. SIEVERT (Sweden)

1937, Chicago

G. W. C. KAYE, Hon. Secretary (Great Britain)
H. BEHNKEN (Germany)
E. PUGNO-VANONI (Italy)
I. SOLOMON (France)
L. S. TAYLOR (United States)

1934, Zurich

R. LEDOUX-LEBARD, Hon. Chairman (France)
F. TANK, 1st Hon. Chairman (Switzerland)
R. BÄR, 2nd Hon. Chairman (Switzerland)
G. W. C. KAYE, Hon. Secretary (Great Britain)
H. BEHNKEN (Germany)
R. SIEVERT (Sweden)
I. SOLOMON (France)
L. S. TAYLOR (United States)
E. PUGNO-VANONI (Italy)

1931, Paris

R. LEDOUX-LEBARD, President (France)
R. SIEVERT, Hon. President (Sweden)
G. W. C. KAYE, Hon. Secretary (Great Britain)
S. MELVILLE, Hon. Secretary (Great Britain)
G. GROSSMAN (Germany)
E. PUGNO-VANONI (Italy)
I. SOLOMON (France)
L. S. TAYLOR (United States)

1928, Stockholm

G. W. C. KAYE, Hon. Secretary (Great Britain)
S. MELVILLE, Hon. Secretary (Great Britain)
L. S. TAYLOR (United States)
G. GROSSMAN (Germany)
R. SIEVERT (Sweden)
I. SOLOMON (France)

3. *Major Steps in the development of protection standards*[12]

1925 ICRU established by First International Congress of Radiology. Maximum permissible dose = 0.1 erythema dose/year = 50–100 r/year (only visible signs of radiation reaction).

1928 ICRP established by Second International Congress of Radiology
- (a) First set of international protection recommendations adopted.
- (b) First unit of dose adopted: the roentgen (r).

1929 NCRP formed in the U.S.

1931 X-ray protection rules produced by NCRP (H 15).

1934 ICRP:
- (a) Adopted first permissible dose of 0.2 r/day (= 72 r/year).
- (b) Based on erythema dose.

1935 NCRP:
- (a) Adopted value of 0.1 r/day for maximum permissible dose (MPD) (= 36 r/year).
- (b) More conservative MPD adopted on basis of their suspected other effects.
- (c) First discussions on genetic effects.

1947 NCRP:
- (a) Lowered MPD to 0.3 r/week (= 15 r/year).
- (b) Lowered because of animal experience in Manhattan District studies.
- (c) Genetic effects considered (see H 59).
- (d) Longevity effects considered.
- (e) Large population effects considered.
- (f) Standards still based mainly on absence of detectable injury to individuals.

1950 ICRP:
- (a) Adopted MPD = 0.3 r/week.
- (b) Used values of NCRP for standards for external irradiation.
- (c) Used unpublished values of NCRP for internal emitters (see H 47).

1952 ICRP/ICRU:
- (a) Held first conference on genetic effects.
- (b) Reached conclusions essentially as recommended by ICRP in April 1956 and NAS in June 1956.
- (c) Felt there was no need to rush to lower levels—better to wait for better data.

1953 NCRP—listed maximum permissible concentration MPC for 100 radioisotopes (H 52).

1953 ICRP:

(a) Adopted NCRP listings of MPC for 100 radioisotopes.

(b) Recommended 1/10 levels for large population groups.

1953 ICRU—adopted unit of absorbed dose (rad).

April 1956 ICRP—Introduced accumulated dose concept for both occupational and whole population exposure.

June 1956 NAS/BMRC—made same general recommendations as ICRP (Natural in view of substantial overlap of membership).

January 1957 NCRP:

(a) Formulated the age-prorated occupational dose to insure proper distribution with age.

(b) Population dose concept of ICRP adopted.

April 1957 ICRP/ICRU:

(a) In work for UNSC made recommendations on population exposure evaluations by sampling techniques.

(b) Tentatively decided that somatic damage is more immediately important than genetic damage.

March, 1958 ICRP:

Met in New York to clarify April 1956 recommendations and prepare final report.

October 1957

Health Physics Pergamon Press 1958. Vol. 1, pp. 105–124

POTENTIAL HAZARD OF WORLD-WIDE Sr^{90} FALLOUT FROM NUCLEAR WEAPONS TESTING*†

WRIGHT H. LANGHAM

Los Alamos Scientific Laboratory, University of California, Los Alamos, New Mexico

(*Received* 13 *December* 1957)

Abstract—Because of the chemical similarity of strontium and calcium, Sr^{90} from weapons tests has found its way into the biosphere and into the bones of man. Two theories as to the mechanism of world-wide fallout have been proposed. The LIBBY theory proposes rapid stratospheric mixing with uniform leakage back into the troposphere where the fallout is deposited according to meteorological patterns. The MACHTA theory proposes slow stratospheric mixing with unequal leakage back from the stratosphere as a result of tropospheric-stratospheric air mixing in the vicinity of the jet streams. Surface build-up in the population belts near 40°N and 40°S latitude may be higher on the basis of the MACHTA theory. At present, surface concentrations of Sr^{90} in the northern United States is approaching 30 mc/mile^2. On the basis of ecological data, soil analyses, and bone analyses, present average maximum equilibrium bone levels in the northern United States is estimated at about 2 pc/g Ca. Assuming no more weapons tests, these levels may reach 3–4 pc/g Ca. Because of inequalities in soil calcium, biological variations, etc., a few individuals may approach bone levels three to five times the average. If biospheric Sr^{90} contamination continues indefinitely at the rate indicated during the past 5 years, average equilibrium levels may increase by a factor of about 10. In the event of war, exploding 20,000 MT equivalents of fission may bring the average maximum equilibrium bone levels in the world population belts to 1000 pc/g Ca, the maximum permissible level for atomic workers. Two thousand megaton equivalents of fission would bring the average maximum equilibrium level to about 100 pc/g Ca, the maximum permissible level for large areas of population. The possible significance of present and future Sr^{90} bone levels is discussed in terms of world health using, as a basis, the life-time natural background radiation to the bone. Much of the present controversy over the significance of Sr^{90} fallout results from disagreement over whether such chronic effects as leukemia and bone sarcoma are threshold or nonthreshold phenomena and over what is an acceptable maximum permissible Sr^{90} bone level for the world population. Present Sr^{90} levels in the population are not great enough to be of any significance to world health. Whether weapons tests should or should not be stopped depends on the importance of maintenance of a balance in nuclear weapons in averting nuclear war. This answer cannot be derived from Sr^{90} fallout data.

INTRODUCTION

DURING the past few years public attention has been increasingly focused on the potential hazard to the general population of widespread, low-level, radioactive fallout from nuclear weapons testing.[1–6] Fission products from bomb tests are being deposited over the surface of the earth, increasing the external gamma radiation background and finding their way into the human body through direct fallout on vegetation and by transmission along the ecological cycle from soils-to-plants-to-animals and to man. Increase in external radiation exposure may result in an increase in incidence of genetic changes. Increased internal radiation

* Work done under the auspices of the U.S. Atomic Energy Commission.

† A large part of the material in this report was presented before the Hearings of the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, 27–29 May, 3–7 June, 1957.

Table 1. Potentially hazardous radionuclides in distant fallout from nuclear detonations

Radioelement	Type radiation	Fission abundance* (%)	Radiological half-life	Absorption on ingestion (%)	MPL (μc)
Pu^{239}	α	—	24,000 yr	3×10^{-3}	0.037
Cs^{137}	β, γ	6.2	26.6 yr	100	54
Sr^{90}–Y^{90}	β	5.1	27.7 yr	30	1
Pm^{147}	β	2.6	2.6 yr	1×10^{-2}	60
Ru-Rh^{106}	β, γ	0.5	1.0 yr	3	3
Ce-Pr^{144}	β, γ	5.3	285 days	1×10^{-2}	5
Zr^{95}	β, γ	6.4	65 days	1×10^{-2}	26
Y^{91}	β	5.9	58 days	1×10^{-2}	5
Sr^{89}	β	4.6	51 days	30	4
Nb^{95}	β, γ	6.4	35 days	1×10^{-2}	76
Ba–La^{140}	β, γ	6.0	12.8 days	5	4
I^{131}	β, γ	2.8	8 days	100	0.7

* Slow neutron fission of U^{238}, abundance in weapon debris is somewhat different.

exposure may cause a greater incidence of somatic diseases in the tissues in which the various fission products deposit upon entering the body.

As a potential internal hazard, Sr^{90} is believed to be the most important radionuclide. Its similarity to calcium, long physical and biological half-life, and high relative fission yield lead to high incorporation in the biosphere and a long residence time in bone. World-wide contamination will result in the bones of the population eventually reaching an equilibrium state with Sr^{90} in the environment. The predominance of Sr^{90} over other long-lived radioelements as a potential hazard can be seen from the data in Table 1, which show that it is the only isotope that combines high fission yield, long half-life, high absorption rate, and a low maximum permissible level (MPL). These data suggest Cs^{137} as the second most important fission product, and its presence in people and foodstuffs has been reported.[7,8] Its potential hazard to the population, however, is believed to be less than that of Sr^{90} by at least an order of magnitude.[8]

Appraisal of the potential hazard from world-wide Sr^{90} fallout requires consideration of the extent and rate of fallout, its mode of incorporation into the biosphere and the human body, present and predicted levels in soils and people, the basis of presently accepted maximum permissible body levels, and the biological significance of present and future levels in terms of the megatons of fission weapons detonated.

DISTRIBUTION OF FALLOUT FROM NUCLEAR DETONATIONS

LIBBY[2,3] was first to propose a model explaining fallout and distribution of atomic debris from nuclear weapons detonations. His model clearly recognized the existence of three kinds of fallout—local, tropospheric and stratospheric.

Local fallout is deposited in the immediate environs of the explosion during the first few hours. This debris consists of the large particles from the fireball and includes partially or completely vaporized residues from the soil and structures which are swept into the cloud. The fraction of total radioactivity which falls out locally is largely dependent on those firing conditions which govern the amounts of soil and extraneous debris incorporated in the fireball. Local fallout from tests is of little concern since it is confined to uninhabited areas and does not get widely disseminated in the biosphere. In the event of war, local fallout from surface and subsurface detonations of weapons with high fission yields could produce

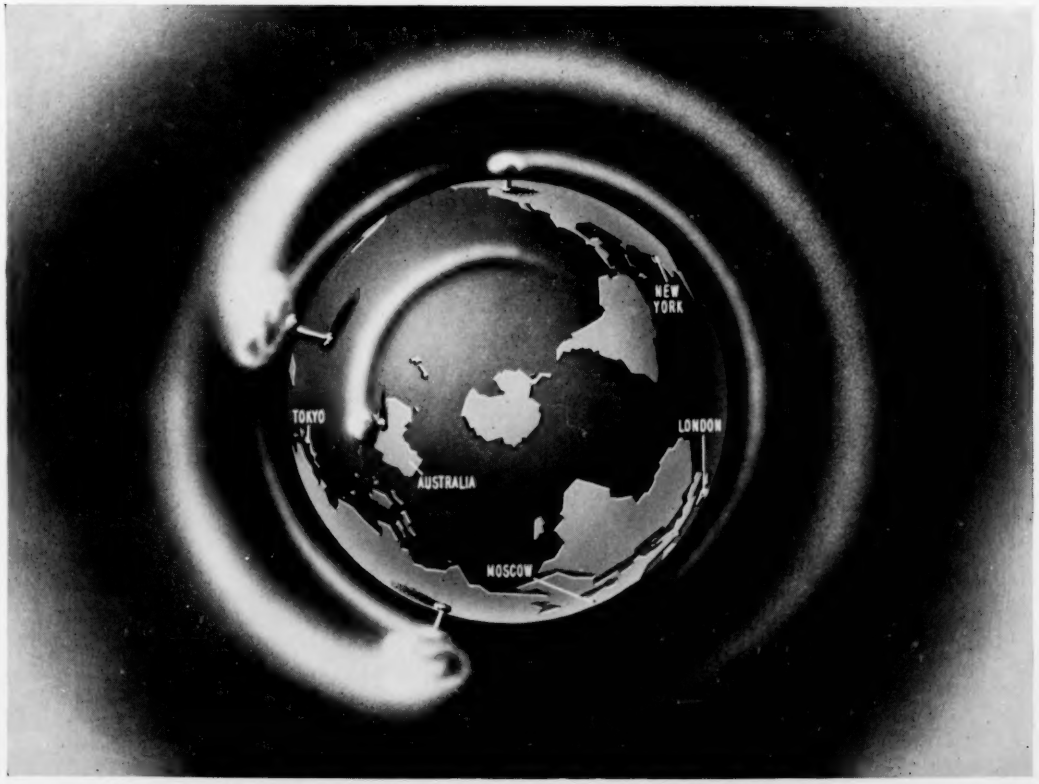


FIG. 1. Local, tropospheric, and stratospheric fallout in relation to weapons test sites.

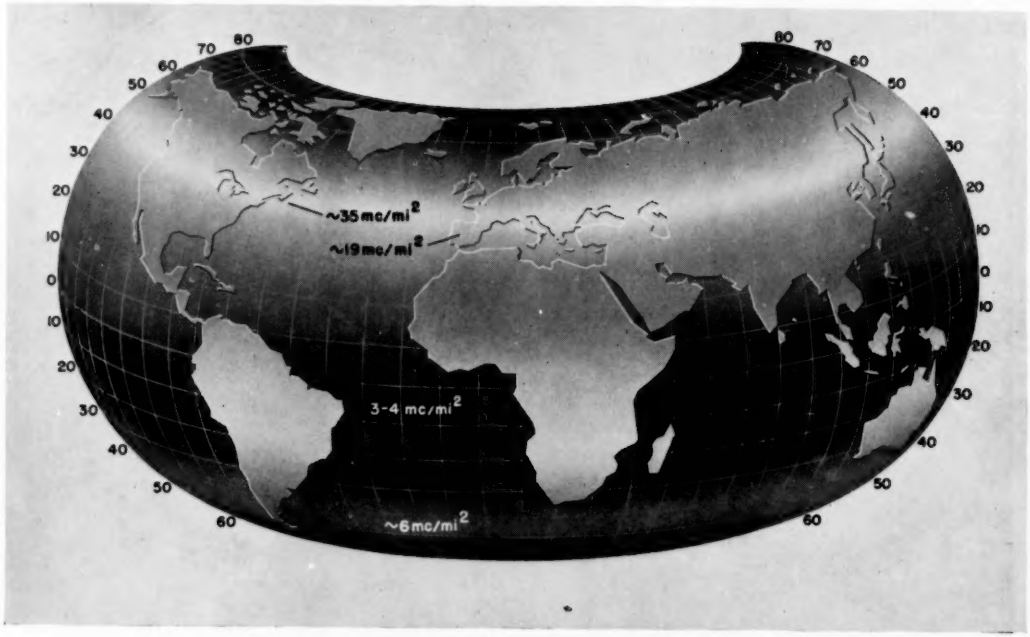


FIG. 2. World distribution pattern of radioactive fallout.

serious contamination of large areas with short and intermediate lived radionuclides.

Tropospheric fallout consists of that material injected into the atmosphere below the tropopause which is not coarse enough to fall out locally. This debris is sufficiently fine that it travels great distances, circling the earth from west to east in the general latitude of the explosion, until removed from the atmosphere by rain, fog, contact with vegetation, and other meteorological and/or physical factors. The half-time of tropospheric fallout is estimated at 20 to 30 days. The fraction of fallout in this category depends mainly on size of the explosion and conditions of firing. If the explosion exceeds a certain minimum size (about 1 MT, i.e. equivalent in energy released to 1,000,000 tons of TNT), the cloud will have enough energy to penetrate the tropopause, carrying fission products into the stratosphere. Smaller detonations leave in the troposphere all debris not deposited locally. The fraction of fission products from a large weapon that remains in the tropopause depends on the size of the explosion, conditions of firing, and meteorological factors.

Stratospheric fallout is composed of the fission products that are carried above the tropopause and can result only from large weapons. LIBBY[9] has postulated that atomic debris, once it is injected above the tropopause is mixed rapidly throughout the stratosphere and falls back uniformly into the troposphere with a half-time of about 7 years. As it returns to the troposphere, it is deposited over the earth's surface in relation to meteorological conditions. The fraction of the fission products deposited in the stratosphere depends on the total energy release of the weapon, latitude of detonation, and conditions of firing.

Local, tropospheric, and stratospheric fallout are illustrated in Fig. 1 in relation to four of the world's five present weapons test sites.

On the basis of his model postulating rapid stratospheric mixing and uniform leakage into the tropopause, LIBBY attributed higher Sr^{90} soil concentrations in the United States to meteorological conditions and to local and tropospheric fallout as a result of the proximity of the Nevada test site. He attributed the higher conditions in the north temperate latitudes to prevailing meteorological conditions and their effects on tropospheric fallout from tests in the U.S.S.R. and at the United States Pacific proving grounds.

MACHTA, in testimony given during the hearing of the Joint Committee of Atomic Energy,[10] proposed a model of stratospheric fallout which differs in some respects from LIBBY's. He postulated that stratospheric mixing is slow and that stratospheric distribution of fission products is still nonuniform. He feels that a major portion of the nuclear debris is still in the northern portions of the northern hemisphere, rather than uniformly spread over the entire globe or even uniformly dispersed in the northern hemisphere itself. He feels also that stratospheric movement of the fission products is largely by direct transport from west to east in the general latitude of the point of injection with very slow vertical mixing. Slow polewards circulation of stratospheric air from equatorial regions provides some mixing towards the poles. The higher concentration of fallout in the temperate latitudes is explained on the basis of air exchange between the troposphere and stratosphere through the break in the tropopause frequently found in the vicinity of the jet stream. A large part of the higher concentration of Sr^{90} found in the northern part of the United States may result from preferential stratospheric leakage in the vicinity of 30°N–40°N latitude instead of the proximity of the Nevada test site.

Qualitatively, either model predicts a world-wide Sr^{90} distribution pattern like that shown in Fig. 2. Quantitatively, the MACHTA model predicts a greater degree of nonuniformity of fallout over the earth with higher deposition of fission products in the north and south temperate latitudes from nuclear debris still in the stratospheric reservoir.

AVERAGE MAXIMUM SURFACE DEPOSITION LEVELS

Present levels

The New York Health and Safety Laboratory has extensive programs under way to study the world-wide distribution of radioactive fallout.[11]

Soil analyses provide the most direct means of obtaining total accumulated Sr^{90} fallout. Unfortunately, the analysis is extremely difficult and too few samples have been analyzed to give a satisfactory picture of geographic distribution. A crude indication, however, of latitudinal distribution of integrated Sr^{90} fallout as of June 1956, derived from soil data, is shown by the lower curve in Fig. 3. This curve is

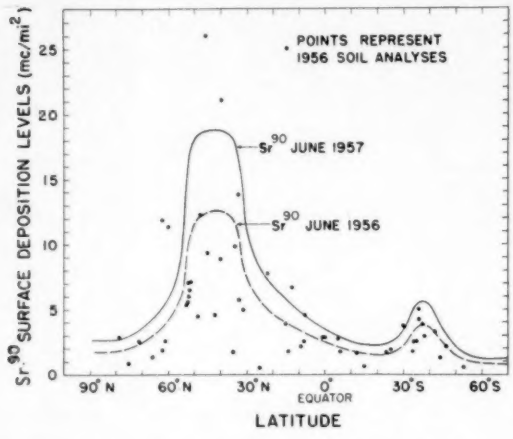


FIG. 3. Integrated Sr^{90} soil deposition levels.

essentially the same as the one given by MACHTA[10] except a few points have been added and the peak concentration in the north temperate latitudes is drawn slightly higher to allow some weighting for average Sr^{90} levels in United States soils. These data suggest a level of about 13 mc/mile^2 for the north temperate latitudes. No soil data are available yet for mid-1957. Fallout data from pot collections in New York and Pittsburgh, however, showed that cumulative Sr^{90} fallout increased by about 50 per cent from June 1956 to June 1957.[12] The upper curve in Fig. 3 represents latitudinal fallout distribution in June 1957. Some of the increase in New York and Pittsburgh fallout could have been tropospheric contribution from Russian tests, which would result in over-prediction of the Sr^{90} levels in other areas. This and other criticisms, however, seem minor compared to the uncertainty in the primary soil data.

Estimated deposition levels in June 1957 show a total Sr^{90} fallout of about 19 mc/mile^2 for the north temperate latitudes, 2–3 mc/mile^2 for the equatorial regions, and about 5.6 mc/mile^2 for the south temperate latitudes. Data from pot collections in the New York area suggest total Sr^{90} deposition levels of about 35 mc/mile^2 in the northern United States in mid-1957. The rapid-build up of Sr^{90} in the northern states in the spring of 1957 cannot be attributed to tropospheric fallout from Nevada tests since Operation Plumbbob had not begun. It may be due to tropospheric fallout from spring test operations in the U.S.S.R. and to preferential stratospheric fallout from past tests.

The total amount of Sr^{90} deposited over the earth's surface (from both tropospheric and stratospheric fallout) as of mid-1957 can be estimated from the upper curve in Fig. 3 by replotting the data in terms of Sr^{90} deposition/degree times the earth's area/degree. This calculation suggests a world total deposition of 1.64 Mc, which gives a world average surface level of 8.2 mc/mile^2.

LIBBY's[3] estimates of Sr^{90} surface deposition levels for the fall of 1956 were 22 mc/mile^2 for the northern United States, 15–17 mc/mile^2 for similar latitudes elsewhere,* and 3–4 mc/mile^2 for the rest of the world. These values are in good agreement with those estimated from the lower curve of Fig. 3. He also estimated the stratospheric reservoir at about 2.4 Mc (24 MT equivalents of fission yield). Using his deposition values and the world Sr^{90} production up to that time, 24 MT equivalents still in the stratosphere would be possible only provided local fallout from surface detonation of megaton weapons was about 25 per cent. His predictions were based on 1955 soil analyses, his model of tropospheric and stratospheric fallout, and a general knowledge of the megatons of fission devices detonated during the spring and summer of 1956.

Values for the world total production (about 5.5 Mc to mid-1957) and deposition of Sr^{90} may be used to estimate the present magnitude of the stratospheric reservoir. Fallout measurements from United States megaton detonations in the Pacific suggest that 50 ± 17 per cent of fission debris falls out locally. The rest is partitioned between tropospheric and stratospheric fallout. Since the fallout time of tropospheric debris is of the order of 20–30 days, the material not accounted for by local fallout plus total world-wide deposition must still be in the stratospheric reservoir. Such a material balance calculation estimates the stratospheric Sr^{90} content at 1.11 ± 0.93 Mc, or the equivalent of 11.1 ± 9.3 MT of fission yield. The average is about one-half the value estimated by LIBBY.[3]

Information on world-wide distribution of Sr^{90} is summarized in Table 2. The fallout levels given for

* The north temperate fallout band was indirectly defined as the region between 60°N to 10°N latitude. It is assumed that the surface deposition of 16 to 17 mc/mile^2 applies to this area.

Table 2. Summary of world-wide distribution of Sr^{90}

Region	Levels of Sr^{90}	
	October 1956	June 1957
Northern U.S.A.	26 mc/mi²*	35 mc/mi²‡
North temperate latitude	13–17 mc/mi²†	19 mc/mi²§
South temperate latitude	3–4 mc/mi²†	5.6 mc/mi²§
Rest of world	2–4 mc/mi²†	3–4 mc/mi²†
World average	6–8 mc/mi²†	8.2 mc/mi²§
Total world distribution	1.13–1.62 Mc†	1.64 Mc§
Stratospheric	1.11–2.4Mc†	1.1 ± 0.9 Mc§

* From U.S.A. soil analyses.[11]

† From this report and LIBBY.[3] Mc represents megacuries.

‡ Health and Safety Laboratory, New York pot collection data.[12]

§ This report.

the various areas are general averages only and assume uniform distribution within the respective areas. Actually, this general picture is greatly oversimplified. Once fission products are suspended in the troposphere (either directly from the detonation or from stratospheric leakage, regardless of mechanism) meteorological conditions play a major role in their surface distribution. LIBBY has stressed the importance of rainfall, snow, fog, and mist.[3] Within any major area fluctuations in levels of surface deposition may occur which correlate with local meteorological conditions. These fluctuations may be expected to correlate better with frequency of rainfall, etc., than with total precipitation. MACHTA[10] has guessed that areas as large as milk-sheds may not have more than two to three times the average deposition for the latitude. He points out, however, that desert areas where there is practically no rainfall may have almost zero fallout.

Future levels (assuming no more tests)

Fallout of Sr^{90} from the stratospheric reservoir will continue even if weapons tests are stopped. Whether the integrated surface deposition levels continue to build up will depend on whether the rate of fallout from the stratosphere more than compensates for the rate of decay of the Sr^{90} already on the ground.

From the surface deposition levels in Table 2 and the value of 1.1 ± 0.9 Mc for the stratospheric reservoir, it is possible to estimate future deposition levels assuming no more weapons tests.

If $M(t)$ is the surface deposition level and $Q(t)$ is the stratospheric storage in mc/mile² at any time, the rate of change of the surface deposition level is:

$$\frac{dM(t)}{dt} = -\lambda M(t) + kQ(t)$$

where λ is the radioactive decay constant of Sr^{90}, and k is the stratospheric fallout rate constant (assumed to be first order). If $\lambda M(t) = kQ(t)$, dM/dt is zero. In this case, additional stratospheric fallout just compensates for radioactive decay, and $M(t)$ does not change. Such an equilibrium state is transitory, since $Q(t)$ is constantly decreasing (both by decay and by fallout). Loss by radioactive decay in $M(t)$, therefore, soon exceeds gain from $Q(t)$, and $M(t)$, falls. If λM_0 is greater than kQ_0 (where M_0 and Q_0 are the concentrations at $t = 0$, the time of cessation of tests), the latter situation already exists and the ground level begins to fall when testing stops. Only if λM_0 is less than kQ_0 will additional fallout from the stratosphere exceed the decay of the ground contamination and the surface deposition level continue to rise. If the mean times of decay and fallout are 40 and 10 years, respectively, $Q(t)$ must be at least $\frac{1}{4}M(t)$ for surface deposition to increase.

Predicted future levels calculated on this basis, assuming uniform and nonuniform stratospheric fallout, are given in Table 3. In the case of nonuniform fallout, it is assumed that future fallout will have the same degree of nonuniformity indicated by present soil levels, that is, the effective stratospheric storage, $(Q(t)_e)$ for a given area is related to the average stratospheric storage $(Q(t)_{av})$ by the equation:

$$Q(t)_e = \frac{M(t)}{M(t)_{av}} Q(t)_{av}$$

where $M(t)$ is the observed ground concentration in the area in question, and $M(t)_{av}$ is the averaged world-wide ground concentration. On the basis of this assumption, the soil levels increase everywhere by the same ratio and reach a maximum about 1963, which is some 10 per cent higher than present levels.

Assuming uniform stratospheric fallout, some areas do not increase since the additional stratospheric fallout is insufficient to compensate for radioactive decay. The time of maximum ground concentration (where it does occur) also varies with location, being about 1966 in the south temperate latitudes and 1969 elsewhere.

Neither method of estimation is strictly correct. The assumption of uniform fallout may underestimate build-up in the northern latitudes, and the assumption of nonuniformity of future fallout according to the past will tend to overestimate build-up in those areas

Table 3. Predicted average maximum surface deposition levels (assuming no more weapons tests)

Region	Present levels June 1957 (mc/mi²)	Future levels	
		Uniform fallout 1967* (mc/mi²)	Nonuniform fallout 1963* (mc/mi²)
Northern U.S.A.	35	30†	39
North temperate latitude	19	17†	21
South temperate latitude	5.6	7	6
Rest of world	3.5	5.2	4
World average	8.2	8.8	9

* Approximate date of maximum when it occurs.
† Decreasing surface deposition level, $\lambda M_0 > kQ_0$.

where some of the material deposited in the past did not come from stratospheric fallout alone. As was stated by MACHTA,[10] it is hoped the truth lies somewhere in between. It must also be kept in mind that the stratospheric reservoir may well be 2.4 Mc as estimated by LIBBY.[3]

Future levels (with continued testing)

Little information is available on rate of biospheric Sr^{90} contamination from weapons test operations. Statements before the Congressional Hearings[6] assumed a constant test rate of 10 MT of fission yield per year, beginning in the spring of 1952, which leads to a world total production of about 5.5 Mc of Sr^{90} to mid-1957. The estimate of total production may be reasonably correct; however, the assumption of a uniform test rate is obviously not correct since a large fraction of the total Sr^{90} production occurred during the United States Castle test series in 1954.

If the testing rate (in terms of fission yield) is held constant, the rate of decay of Sr^{90} in the biosphere will eventually equal the rate of production, and continued testing will result in no further increase in deposition levels. This equilibrium will be reached in about 100 years.

CAMPBELL[5] and STEWART *et al.*[13] have estimated surface deposition levels at equilibrium with a uniform test rate, and their calculations suggest levels about thirty times the present. Their equation was derived, however, from stratospheric fallout and applies to ground levels due to the stratospheric component only.

LIBBY[14] estimated surface build-up on the basis of total levels on the ground at $t = 5$ years and predicted equilibrium levels eleven times the present values. His calculations have been checked by others, including NEWMAN.[15] LIBBY also assumed that about 30 per cent of the Sr^{90} (over the long period of time required for equilibrium) would become unavailable to plants and the available equilibrium levels would be only about eight times the present values. Attempts are being made to obtain actual yearly Sr^{90} production rates to further refine predictions of surface levels under continued testing. Until then, an equilibrium build-up factor of about 10 with a continued average test rate of 10 MT of fission yield per year seems reasonable. Table 4 shows the average maximum future surface deposition levels calculated from the data in Table 2, assuming equilibrium with a constant test rate of 10 MT of fission yield per year will result in a build-up factor of 10.

Others have made similar estimates. LIBBY[16] estimated equilibrium surface deposition levels for the United States at 400–600 mc/mile². NEWMAN[15]

Table 4. Average maximum surface deposition levels at equilibrium (assuming a continued testing rate of 10 MT of fission yield per year)

Region	Present levels (mc/mi²)	Equilibrium level (mc/mi²)
United states	35	350
North temperate latitude	19	190
South temperate latitude	5.6	56
Rest of world	3.5	35
World average	8.2	82

estimated a United States deposition level of about 400, and MACHTA[10] 350–850 mc/mile^2.

Incorporation of Sr^{90} *into the Biosphere*

When fission debris is deposited on the earth's surface and incorporated in the soil, the individual nuclides are taken into plants through the root system according to their specific soil-plant relationships. That which settles directly on vegetation may remain as surface contamination or may in part enter the plant through foliate absorption. When plants are eaten by animals, the radioactivity incorporated in the plants (either by foliate absorption or from the soil) or deposited on their surfaces, is absorbed and retained by the animal according to the specific metabolic characteristics of the individual nuclides. When plant and animal products are eaten by man, the radioelements they contain are absorbed and incorporated into his tissues, again in accordance with their individual metabolic properties.

Sr^{90} is chemically and metabolically similar to calcium. Therefore, it is incorporated into the biosphere along the same ecologic chain as is calcium. It is taken into plants through the root system in relation to available soil calcium and absorbed and deposited in human bone in relation to the Sr^{90}/Ca ratio in the diet.

It is reasonable to assume that strontium may be discriminated against with respect to calcium in passing along the ecological chain. For example, the Sr^{90}/Ca ratio of human bones may be expected to be lower than that of soil. Attempts are being made to determine the overall Sr^{90}/Ca discrimination ratio in going from soils to human bone by individually determining the discrimination factors (DF) that occur at the various steps along the ecological chain. The discrimination factor most difficult to establish is the one from soils-to-plants (DF_1). It is dependent, among other things, on type of soil, available soil calcium, type of plant and perhaps on rainfall, all of which may vary greatly with geographic location. MENZEL[17] obtained a soil-to-plant discrimination factor $\{(Sr^{90}/Ca)_{plant}\}/\{(Sr^{90}/Ca)_{soil}\} = 0.7$, for four widely different soil types using both radioactive and stable strontium. LARSON[18] and BOWEN and DYMOND[19] obtained comparable values.

ALEXANDER *et al.*[20] obtained excellent data on the discrimination against strontium over calcium in going from plants-to-milk (DF_2). They compared the stable strontium to calcium ratio of cows' milk with that of the feed the cows consumed and found that $\{(Sr^s/Ca)_{milk}\}/\{(Sr^s/Ca)_{feed}\} = 0.13$. These authors[21] also studied the relative uptake of stable strontium and calcium from the diet by a variety of rodents, including rabbits and kangaroo rats from the Nevada desert. They found an average $\{(Sr^s/Ca)_{bone}\}/\{(Sr^s/Ca)_{food}\} = 0.24$.

COMAR[22] determined the discrimination factor from plants-to-milk (DF_2) using Sr^{85} and Ca^{45} and found that $\{(Sr/Ca)_{milk}\}/\{(Sr/Ca)_{plants}\} = 0.14$, which is in good agreement with the value of 0.13 obtained from stable strontium data.[20] He also studied the discrimination factor (DF_3) in going from milk-to-human bone following single and multiple feeding and found the ratio $\{(Sr/Ca)_{bone}\}/\{(Sr/Ca)_{milk}\} = 0.5$, and that the ratio was influenced by various constituents of the milk.

LASZLO *et al.*[23] and COMAR disagree as to the discrimination against Sr^{90} in going from milk-to-bone, since the former author feels that the discrimination factor against Sr^{90}, regardless of dietary source, is about 0.25. KULP *et al.*[24] have recently reported data which support the lower value. Experiments by LASZLO[25] on the discrimination factor (DF_4) from plants directly to human bone gave a value of 0.25, which agrees with radioisotope studies in rats conducted by COMAR[26] and with stable strontium data on rodents.[21]

The overall ratio ($OR_{bone-soil}$) in going from soil-to-human bone via the diet may be estimated from the various discrimination factors and the fraction of dietary calcium derived from dairy products and from other sources. For the United States population the amount of dietary calcium derived from dairy products is estimated at about 80 per cent. The remainder is derived from cereals, vegetables, meats, etc. On the basis of the above generalizations, $(OR_{bone-soil}) = (0.8 \times DF_1 \times DF_2 \times DF_3) + (0.2 \times DF_1 \times DF_4) = (0.8 \times 0.7 \times 0.13 \times 0.25) + (0.2 \times 0.7 \times 0.25) = 0.05$ and indicates that the equilibrium concentration of Sr^{90} in bone calcium for the United States population will be about 5 per cent of the concentration in the available calcium of the soil. The fraction of dietary calcium derived from dairy products varies widely among the various populations. A general expression for the ecological discrimination factor is

$$(OR_{bone-soil}) = (M_f \times 0.023) + (R_f \times 0.175)$$

in which M_f and R_f are the fractions of dietary calcium derived from dairy products and from other sources, respectively.

It should be emphasized strongly that the expression derived above applies only to passage along the ecological chain. An ecological discrimination factor automatically assumes that calcium and strontium are uniformly mixed in soil to the average depth of the plant feeding zone. No allowance is made for direct foliar absorption of Sr^{90}, for its dilution with a greater reservoir of available soil calcium through

plowing, for the possibility that it may become less available with time through soil binding and leaching, or for differences in uptake by different plant species.

The mode of ecological discrimination against strontium with respect to calcium during passage along the chain from soil-to-human bone via the diet is illustrated in Fig. 4.

Data on stable strontium content of human bone ash were reported by HODGES *et al.*[27] and show conclusively that, under equilibrium conditions, stable strontium is equally distributed throughout the skeleton (Table 5). Their results were confirmed by others,[28,29] and leave little doubt but that man's bones will eventually come into equilibrium with the Sr^{90} contamination in his environment.

Sr^{90} *in Bones of the Population*

Present and future average maximum Sr^{90} equilibrium levels in bones of the population can be estimated from the soil-to-bone discrimination ratio, the ratio of milk to other sources of calcium in the diet, and the present and predicted average maximum surface deposition levels developed in the previous sections.

Assuming an average of 20 g available Ca/ft² of soil to a depth of 2½ in.,* 1 mc of Sr^{90}/mile² is equivalent to 1.8 pc Sr^{90}/g available soil calcium. Assuming all the the Sr^{90} is in available form, multiplication of the surface deposition levels given in Table 2 by 1.8 gives the Sr^{90} activity per gram of available soil calcium. Multiplication of the specific activity of available soil calcium by the Sr^{90} discrimination ratio should give the average maximum specific activity of calcium laid down in the adult skeleton through exchange and bone remodeling during the period of environmental contamination and the average maximum Sr^{90} concentration in a skeleton at equilibrium with the integrated surface deposition levels.

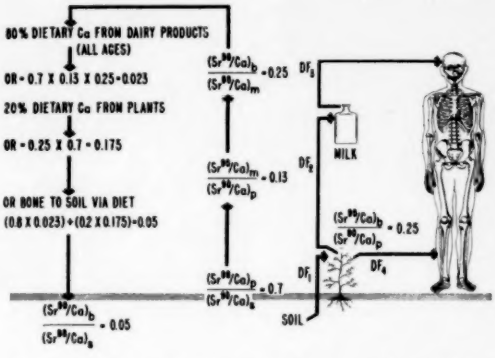

80% DIETARY Ca FROM DAIRY PRODUCTS (ALL AGES)
OR = 0.7 X 0.13 X 0.25 = 0.023
20% DIETARY Ca FROM PLANTS
OR = 0.25 X 0.7 = 0.175
OR BONE TO SOIL VIA DIET
(0.8 X 0.023) + (0.2 X 0.175) = 0.05
(Sr90/Ca)b / (Sr90/Ca)m = 0.25
(Sr90/Ca)m / (Sr90/Ca)p = 0.13
(Sr90/Ca)p / (Sr90/Ca)s = 0.7
(Sr90/Ca)b / (Sr90/Ca)s = 0.05
DF3
DF2
DF1
DF4
MILK
(Sr90/Ca)b / (Sr90/Ca)p = 0.25
SOIL

FIG. 4. Discrimination against strontium with respect to calcium during passage from soil-to-man (United States).

Derivation of Sr^{90}/Ca discrimination ratios $\{OR = (M_f \times 0.023) + (R_f \times 0.175)\}$ from per capita consumption of principal foodstuffs[31,32] and their average calcium content[33] for the various countries of the world was attempted. *OR* varied from about 0.05 for countries with high milk consumption to about 0.15 for those consuming very little milk. Discrimination ratios were weighed for population densities to give weighted average values of 0.095, 0.06, and 0.12 for the north temperate latitudes, south temperate latitudes, and rest of the world population, respectively. Average maximum Sr^{90} equilibrium bone levels in the world's population postulated from ecological considerations are given in Table 6.

Table 5. *Stable strontium content of human bones*

Sample	Strontium in bone ash (%)			
	Parietal	Vertebra	Rib	Femur
Fetal*	0.016	0.016	0.017	0.017
All ages†	0.023	0.022	0.022	0.022
1914 cadavers	0.027	—	0.027	0.025

* Fetal bones showed range of 0.015 to 0.019 per cent and were significantly lower than adult bone due perhaps to a placental discrimination factor.[30]

† "All-age" group showed no significant increase with age when analyses were compared in five age groups.

* This broad, but necessary, generalization was made by LIBBY[2] and based on calcium and Sr^{90} analyses of domestic soils collected in 1953 following the first thermonuclear detonation in the Pacific. These analyses showed that about 56 per cent of the Sr^{90} was in the top 1 in. of soil and the remainder in the next 5 in. The assumption that the effective calcium reservoir with which Sr^{90} is mixed (and the reservoir from which man's dietary calcium is drawn) is the exchangeable calcium in the top 2½ in. of soil is subject to great uncertainty. In cultivated areas, plowing will result in mixing of the Sr^{90} with the exchangeable calcium reservoir of the top 6 in. Leaching over long periods of time will result in removal and mixing with a greater calcium reservoir. It is also possible that Sr^{90} will in time become less available to plants. The uncertainties in this assumption alone are more than adequate to explain any difference in equilibrium bone levels predicted from ecological discrimination factors and from bone analyses. For this reason, it is essential that Sr^{90} levels in the bones of the population be established by more direct methods and not by prediction.

Table 6. Postulated average maximum equilibrium Sr^{90} bone levels in the world population (pc/g *bone* Ca)

Region	Mid-1957		About 1963*		About 2050†	
	Ecol. Data	Bone Data	Ecol. Data	Bone Data	Ecol. Data	Bone Data
United States	3.1	1.7	3.5	1.9	31	17
North temperate latitude	3.2	1.7	3.6	1.9	32	17
South temperate latitude	0.6	0.5	0.7	0.6	6	5
Rest of world	0.8	0.3–0.5	0.9	0.5–0.8	8	3–5
World average‡	(2.8)	(1.5)	(3.1)	(1.7)	(28)	(15)

* Assuming no more weapons tests.
† At equilibrium with a continued test rate of 10 MT equivalents of fission per year.
‡ Population weighted average.

These data are subject to the uncertainties in estimations of surface deposition levels and to the uncertainties in derivation of the bone-to-soil discrimination ratio. The greatest uncertainty probably lies in their dependence on available soil calcium with which the Sr^{90} is mixed. Available soil calcium may vary over the world from about 1 to 100 g/ft^2 to a depth of 2½ in. The relative Sr^{90} uptake would be higher in areas with abnormally low available soil calcium. The available calcium with which the Sr^{90} is actually mixed is dependent also on the average depth of the feeding zone of all the various types of plants responsible for introduction of calcium into man's food chain.

The discrimination ratios for the various areas are only superficially adjusted for differences in population dietary habits and make no allowance for individual variations in calcium metabolism and for the fraction of Sr^{90} entering the food chain through direct fallout on vegetation.

The data in Table 6 suggest the average maximum level of Sr^{90} in the bones of the population of the United States would be about 3.1 pc/g Ca, if they were in equilibrium with the 1957 soil deposition levels. The average of the north temperate population belt would be about the same as the United States because of the low ratio of milk to cereals in the diet of the heavy populated countries of the far east. The world average is only slightly lower than the average for the north temperate latitute, which is not surprising since over 80 per cent of the world's population lives in that region.

LANGHAM and ANDERSON[34] attempted to establish the relationship between fraction of Sr^{90}/Ca skeletal equilibrium as a function of age and rate of biospheric contamination from the rate of skeletal accretion[35]

Table 7. Yearly accretion of skeletal calcium in males

Age (years)	Calcium in total skeleton * (g)	Increase per year (g)	Fraction of equilibrium Sr^{90}/Ca
0	28	—	0.50†
1	100	72	0.79
2	147	47	0.59
3	179	32	0.42
4	201	22	0.29
5	219	18	0.20
6	239	20	0.16
7	264	26	0.16
8	297	33	0.18
9	341	44	0.20
10	396	55	0.22
11	463	67	0.23
12	539	73	0.24
13	624	85	0.23
14	715	91	0.22
15	806	91	0.21
16	894	88	0.19
17	973	79	0.16
18	1035	62	0.13
19	1073	38	0.09
20	1078	5	0.045
21	1078	0	0.019
22	1078	0	0.006
23	1078	0	0.001
24	1078	0	0.000

* From MITCHELL *et al.*[35]
† Assuming 50 per cent fetal protection.[30]

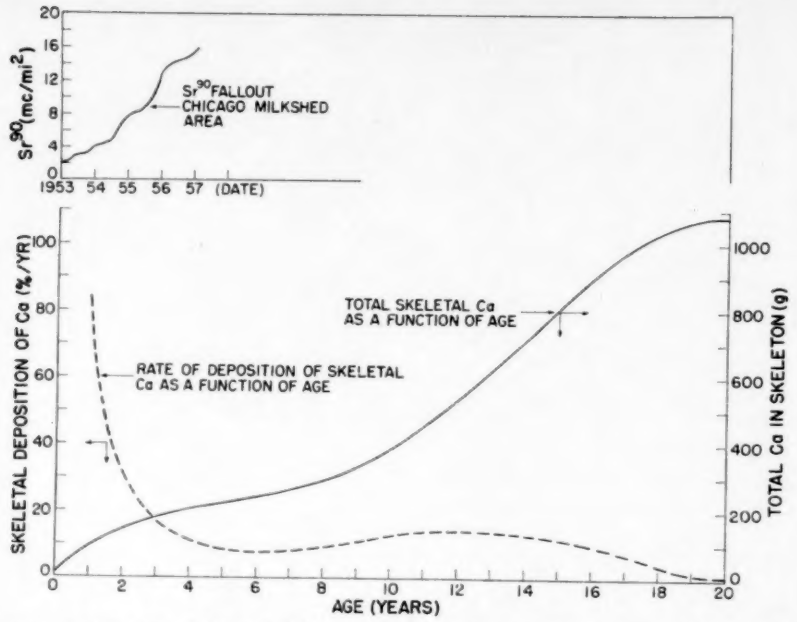

Sr^{90} (mc/mi^2)
Sr^{90} FALLOUT CHICAGO MILKSHED AREA
1953 54 55 56 57 (DATE)
SKELETAL DEPOSITION OF Ca (%/YR)
TOTAL SKELETAL Ca AS A FUNCTION OF AGE
RATE OF DEPOSITION OF SKELETAL Ca AS A FUNCTION OF AGE
TOTAL Ca IN SKELETON (g)
AGE (YEARS)

FIG. 5. Rate of accretion of skeletal calcium in relation to rate of environmental contamination.

(Table 7, Fig. 5), increase in integrated fallout (Fig. 5), and KULP's[36] average Sr^{90} bone values (mostly from north temperate latitudes) for 1955–1956 for the various age groups. The specific activity of the calcium in the bones of children should depend on the fraction of the skeletal calcium laid down by all mechanisms during the period of environmental contamination, and the specific activity of the bone calcium of adults should depend on the per cent of skeletal calcium equilibrated during the same period by new bone formation through bone remodeling plus exchange.

It was assumed that each yearly increment of skeletal growth will contain Sr^{90} at a concentration corresponding to the Sr^{90} build-up in the biosphere for that year. For a first approximation, the skeleton was regarded as a unit and the Sr^{90} burden averaged over the entire skeleton.

Calculated values for the apparent fraction of equilibrium Sr^{90}/Ca ratio as a funtion of age, based on skeletal growth rate alone and a yearly doubling time of the Sr^{90} level, are given in the last column of Table 7 and are shown by the solid curve of Fig. 6. The method of calculation is best explained by an example. For an 8-year old skeleton (Table 7) in 1956, the last 33 g of calcium contain an equilibrium concentration C of strontium, and thus a total amount of strontium equal to 33 times C. The previous year's deposition of 26 g would have been formed with a concentration $C/2$ of strontium and would contribute 13 C. The fifth and fourth years' growth would incorporate 20 $C/4$ and 18 $C/8$ units of strontium, respectively. The total strontium in the skeleton at age eight is, therefore $(33 + 13 + 5 + 2.5)C$, or 53 C. (A 4-year cut-off is used since large-scale testing began in 1952. Actually, the series is converging so

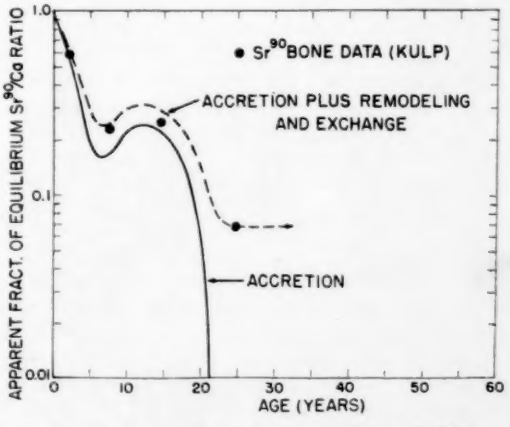


FIG. 6. Apparent fraction of equilibrium Sr^{90}/Ca ratio in relation to age.

rapidly that the cut-off has little effect.) If the entire skeleton had been in equilibrium with the Sr^{90} level of the environment in the eighth year, it would have contained 297 *C* units of Sr^{90}. The fraction of equilibrium is, therefore, 53 *C*/297 *C*, or 0.18.

The points in Fig. 6 represent KULP's 1955–1956 data[36] normalized to the 0- to 4-year age group as representing 59 per cent of equilibrium Sr^{90} concentration.

At age 24 (4 years beyond the age at which skeletal growth stops) these data show that 7 to 10 per cent of the sketetal calcium was involved in bone remodeling plus exchange during the period of environmental contamination. If a similar fraction of the skeletal calcium of growing subjects is involved in exchange plus remodeling, the the Sr^{90} levels in children would be proportionally higher than the curve based on skeletal calcium accretion alone. This indeed appears to be the case and indicates that the major factors have been considered in constructing the model.

If the fraction of remodeling and exchange for children's bones is similar to that observed for adults, then the accretion curve below age 20 should be raised proportionately, to give the overall apparent fraction of equilibrium Sr^{90}/Ca ratio as a function of age (broken line, Fig. 6). This curve permits the use of adequate bone data from any age group to predict the average maximum equilibrium Sr^{90} bone level and indicates an average maximum equilibrium level of 0.9 pc/g Ca at the end of 1955.

Sr^{90} content of skeletons of stillborns[3] during 1955 averaged about 0.5 pc/g Ca, which gives an average maximum equilibrium level of 1.0 when the placental discrimination factor of 0.5[30] is considered. BRYANT *et al.*[37] in England reported analyses of twenty-eight bone samples from subjects of all ages collected about January of 1956. Eight samples from persons ranging from 3 months to $3\frac{1}{2}$ years old (average $1\frac{1}{3}$ years) averaged 0.9 pc Sr^{90}/g Ca, and eleven subjects ranging from 20 to 65 years of age (average 36 years) averaged 0.07 pc Sr^{90}/g Ca (after dividing all rib results by two[36]). The predicted average maximum Sr^{90} equilibrium level about January 1956, based on these age groups, is 1.0 and 0.9 pc/g Ca, respectively.

On the assumption that surface deposition levels had a doubling time of 1 year, LANGHAM and ANDERSON[34] predicted an average maximum bone equilibrium level of 1.8 pc/g Ca for the north temperate latitudes in the fall of 1956. Data on Sr^{90} fallout from pot collection samples in New York and Pittsburgh[12] show, however, that fallout did not double but increased by only about 50 per cent. On this basis, their predicted level for the north temperate population belt in the fall of 1956 would be 1.4 pc/g Ca. KULP[24] applied the same age weighting method to 1956–1957 milk data (assuming a bone-to-diet discrimination ratio of 0.25) and estimated average maximum equilibrium bone levels of 1.1, 0.9, 1.1 and 0.5 for North America, Europe, Asia, and the rest of the world, respectively, by the end of 1956. A crude estimate of future average maximum equilibrium bone levels can be made from the 1956 values by assuming they will be proportional to the average maximum surface deposition levels in Table 3. These estimates are compared in Table 6 with those postulated from ecological considerations. Values postulated from ecological discrimination and from bone analyses differ by a factor of about 2. This discrepancy results mostly from the large discrimination ratios derived for far eastern countries with large populations and low milk consumption. Values from bone data were predicted on the assumption that the analyses represented the average bone levels for the various regions. Bone data are probably more reliable than ecological predictions; however, they may be low since they are probably not weighted adequately for population density because the number of samples from the United States and Europe exceeded the number from countries with a large population and a low relative milk consumption.

Others have estimated average maximum Sr^{90} equilibrium levels in bone, usually by methods that involved surface deposition levels and various ecological and dietary discrimination factors.

LIBBY first estimated the maximum equilibrium level in the United States and Europe in about 1970 at about 11 pc/g Ca.[2] This estimate was based on the assumption that bone calcium would reach a Sr^{90} level approximately 70 per cent of that of available soil calcium. He later estimated a maximum average in the 1970's of 4 to 10 pc/g Ca[3] on the basis that calcium of bone would come into equilibrium at 10 per cent of the Sr^{90} content of available soil calcium. More recently he has estimated the average maximum bone equilibrium level in the spring of 1957 at 1.7 to 3.9 ρc Sr^{90}/g Ca. The latter estimate was based on an overall discrimination ratio of 1/13 to 1/30 for surface deposition levels of 25 mc Sr^{90}/mile^2 between 10° to 60°N latitude. Assuming no more weapons tests, he concluded that the equilibrium bone concentration would be very little higher in the 1970's. If tests were continued indefinitely at the past 5-year rate, he predicted average maximum equilibrium bone levels in the United States of 5 to 20 pc/g Ca.

KULP[36] estimated the world-wide average maximum equilibrium Sr^{90} bone level at 0.6 pc/g in the fall of 1955 and at 1.3 pc/g Ca in about 1970, assuming no more weapons tests after the 1956 series.

His estimates for the United States bone value were 0.9 pc/g Ca for the fall of 1955 and about 2 pc/g Ca by 1970. All estimates were based on Sr^{90} surface deposition levels and dietary discrimination factors. In his most recent paper,[24] he estimated the North American population would reach an equilibrium bone level of about 8 pc/g Ca if weapons tests were continued indefinitely at the past 5-year rate.

EISENBUD[11] estimated the maximum average equilibrium bone level in the New York area (and the northern United States) at about 4–6 pc/g Ca at the time of maximum stratospheric deposition (with no more tests), on the assumption that bone calcium would come into equilibrium with the Sr^{90} of the milk at about 50 per cent of the milk level.

The various estimates of average maximum Sr^{90} equilibrium bone levels show an encouraging trend toward general agreement. The most troublesome feature of such estimates, however, is that they are average maximum equilibrium levels and make little or no allowance for local variations of fallout due to meteorological factors, variations in available soil calcium, dietary patterns and habits, nutritional state of segments of the population, individual metabolic conditions, etc.

Frequency distribution patterns have been reported for stable strontium,[28] natural radium,[38] and Cs^{137}[8] in man. All these nuclides show essentially normal distributions with standard deviations of about 35 per cent. LIBBY[9] has stated (at steady state among people living in a given locality) only one person in about 700 will have more than twice the average Sr^{90} burden, and the chances of anyone having as much as three times the average will be about one in 20,000,000. Presently the Sr^{90} measurements of bone samples from subjects of all ages show a much greater scatter than indicated by a standard deviation of 35 per cent. The greater scatter of the observed values is due laregly to the fact that samples came from many localities and (because of the relatively short period of environmental contamination and the age dependence of Sr^{90} deposition) represent varying degrees of equilibrium conditions. The spread may be expected to decrease as equilibrium is approached.[9,24]

Local meteorological conditions will result in increased intensity of fallout in certain localities. The worst possible situation that could come about would be for these "hot spots" to coincide with localities of low available soil calcium in which the population grew up and lived in provincial isolation. LIBBY[9] has considered this problem in view of the averaging which occurs in food distribution systems and has postulated that a factor of 5 encompasses the total variation due to all factors, including soil calcium deficiencies.

Significance of Sr^{90} *Levels in the Population*

Basis of maximum permissible levels of radiation

Evaluation of the potential hazard of Sr^{90} to the general population requires consideration of the basis of maximum permissible levels of radiation exposure. Human experience through diagnostic and therapeutic use of X- and gamma rays and extensive animal experimentation with all types of radiation have demonstrated conclusively the production of deleterious biological effects. These effects are manifest as early physiologic aging resulting in life shortening, mutations, irregularities in hematopoietic function (some of which result in increased incidence of leukemia), and specific organ or tissue changes such as cataracts of the lens of the eye and tumors of bone.

Damage from ionizing radiations may occur from radioactive isotopes deposited in the tissues and organs, as well as from radiations originating from an external source.

Occupational exposure. Maximum permissible levels for most radioisotopes applied to occupational exposure are calculated on the premise that no critical organ or tissue will receive an average dose rate greater than 300 mrem per week (the maximum permissible dose rate for external whole body radiation), assuming uniform distribution of the isotope throughout the tissue or organ.[39,40] It is recognized that calculations based on average dose or uniform distribution in a critical organ or tissue may lead to considerable error, since some radioisotopes are unevenly deposited. Although the average dose rate to a critical organ may be 300 mrem per week, some portions of the organ may receive considerably less than the average and others correspondingly more. This error, however, is believed to be offset at least in part by the fact that 300 mrem is considered the acceptable weekly exposure to the entire body or the blood-forming organs, and therefore may be overly conservative when applied to a small element of tissue. The integrated yearly dose limitation of 5 rem applies to internally-deposited isotopes only if the critical organ is the

gonads or it constitutes a major part of the total body.

The maximum permissible levels of radionuclides that localize in bone (i.e. Sr^{90}) are determined by direct comparison with the 0.1 μc maximum permissible burden for radium. Limited human experience has indicated conclusively that small amounts of radium fixed in the skeleton will produce osteoporosis, necrosis, and sarcoma.[41] Bone changes in radium dial painters and persons who received radium therapeutically provide the basis for the value of 0.1 μc. The maximum permissible levels of other bone-seeking radioisotopes are established on the premise that the amount fixed in the bone will not result in greater probability of biological effect than that produced by 0.1 μc of fixed radium.

Derivations of formulae for the calculation of maximum permissible levels, their parameters and pertinent information on individual nuclides are given in the Handbooks of the International[39] and the National[40] Commissions on Radiological Protection.

Nonoccupational exposure. Maximum permissible levels for nonoccuptaional exposure or exposure of a large segment of the general population were established by taking arbitrarily one-tenth of the value for working personnel.[39,40,42]

The rationale behind a lower value for the general population is based on the numbers involved in the two groups at risk and the increased heterogeneity of the general population over that of a select working group. On the assumption that frequency of response to radiation stress follows a Gaussian distribution, the probability of injury of a few individuals from a specified dose increases with increase in size and heterogeneity of the group at risk.

The maximum permissible level of Sr^{90} for workers is presently 1 μc in the total adult skeleton, and the recommended level for the general population is 0.1 μc.*[43,44,45] The permissible levels of radiation exposure (including that from Sr^{90}) is predicated on the assumption that chronic and delayed effects of radiation are threshold phenomena (Fig. 7).

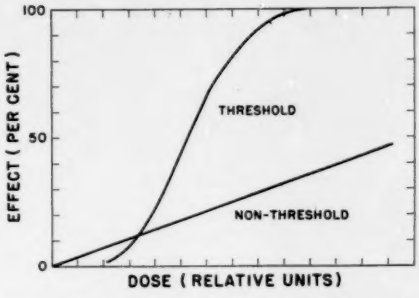


FIG. 7. Threshold and nonthreshold response as a function of radiation dose.

That is to say, there is a threshold dose below which effect essentially becomes zero and above which it increases exponentially over a limited dose range.

Threshold versus nonthreshold response

If chronic effects of radiation are threshold phenomena, 100 pc Sr^{90}/g Ca must be looked upon as a true maximum permissible level and not as an average for large segments of the population. If chronic effects are nonthreshold phenomena and linear with dose, (Fig. 7), the maximum permissible level of Sr^{90} in the bones may be expressed in terms of a population or group average and a portion of the natural population incidence of the effect in question must be attributed to natural background radiation. In this case, the potential hazard should be established on the basis of probability of risk averaged over the entire population or group.

At present it is impossible to say whether leukemogenic and sarcogenic responses to chronic radiation dosage are threshold or nonthreshold relationships. The recent Congressional Hearings[6] failed to produce any degree of unanimity of opinion among the experts. Argument for a linear relationship between incidence of leukemia and radiation dose was presented recently by LEWIS.[46] His argument was based on all major sources of human data and included a consideration of the

* There is about 1 kg of calcium in the adult human skeleton; therefore, the MPL of Sr^{90} in the general population is equivalent to 0.1 μc Sr^{90}/kg Ca = 100 mμc/kg Ca = 100 pc/g Ca = 100 sunshine units.[3]

Japanese atomic bomb survivors, the British cases of X-ray treated spondylitis patients, X-ray treated cases of thymic enlargement, practising radiologists, and spontaneous incidence of leukemia in Brooklyn, New York. The validity of the conclusion was questioned by WARREN, BRUES, and others during the Congressional Hearings.[6] Radiation as a carcinogenic agent has been discussed at length by BRUES,[47] who stated that the relation between radiation dose and carcinogenic effect is not easy to find and a critical experiment has yet to be done which will clearly indicate, even in a single instance, what the relation is over more than a small range of dosages. While admitting that it is not known, he proposes that a threshold relationship between radiation dose and tumor incidence does exist.[48]

Hazard from present and predicted Sr^{90} *levels*

The significance of the general hazard of present and predicted levels of Sr^{90} in bone can be evaluated only in relation to human experience, which is indeed inadequate. Bone sarcoma has resulted from a fixed skeletal burden of 3.6 μc of pure Ra^{226}, and nondeleterious bone changes have been observed in persons having only 0.4 μc for a period of 25 years.[49] Necrosis and tumors of the bone have occurred also several years after large doses of X-ray,[50] and consideration of human experience with leukemogenic effects of X- and γ-radiation[43,46,51] suggests that about 80 rads may double the incidence of leukemia.

The only other human experience with which present and predicted levels of Sr^{90} may be compared is that arising from natural background radiation. Natural background dose to the bone (during a 70-year lifetime) may vary from about 8 to 38 rems.[52] The major contribution to background variation is differences in the radium levels of soils and minerals. The average natural skeletal radiation dose rate was carefully evaluated by DUDLEY and EVANS[53] and their data are shown in Table 8.

Fig. 8 gives a general summary of estimated skeletal radiation doses from accepted maximum permissible levels and from present and predicted Sr^{90} burdens in relation to human experience. The maximum permissible level of Sr^{90} (100 pc/g Ca) is estimated to deliver about 8.5 rads* to the skeleton during a 70-year lifetime. This is comparable to the average natural background dose to the bone for the same time period, and a factor of ~4 below the maximum natural background dose to which small segments of the general population may be exposed, as a result of differences in altitude and natural radium content of soils and minerals. It is a factor of 40 below the lowest skeletal dose which has produced minimal nondeleterious bone changes. These data suggest that the present average maximum Sr^{90} equilibrium level will result in a lifetime radiation dose of approximately 2 per cent of the accepted maximum permissible level for the general population. With continued biospheric contamination indefinitely at the past 5-year rate, the average maximum radiation dose may approach about 20 per cent of the presently accepted maximum permissible level.

Table 8. *Average natural background radiation to the skeleton*[53]

Source of radiation	Skeletal dose rate (mrem/year)	Total dose to age 70 (rem)
K^{40} (internal)	8	0.56
Ra^{226} (internal)	12	0.84
MsTH (internal)	12	0.84
RaD (internal)	12	0.84
Cosmic rays (external)	30	2.1
Local γ-rays (external)	60	4.2
Total	134	9.4

The biological significance of present and predicted Sr^{90} average maximum equilibrium levels and maximum permissible levels for occupational and nonoccupational exposure is summarized in Table 9.

Without scientific justification, but to present

* A dose of $8\frac{5}{10}$ rads is the calculated one assuming incorporation to age 20 and decay to age 70 with no more incorporation. If equilibrium were maintained, the calculated skeletal dose would be about 21 rads. Since some but not all of the skeleton undergoes remodeling plus exchange, somewhere between 8 and 21 rads is probably more correct.

the worst possible potential hazard from Sr^{90} biospheric contamination, a comparison may be made between the radiation dose from present Sr^{90} bone levels and the postulated leukemia doubling dose.[46] Assuming a nonthreshold response and that 10 per cent of the natural incidence of leukemia in the population (6/100,000) is a result of natural background radiation, the average maximum Sr^{90} equilibrium bone level for the north temperate population belt would be equivalent to about 1.7 leukemia cases per 10,000,000 population. Averaged over the world population of 2.6 billion, this would produce an increased leukemia burden of 400 cases per year. A world average of 100 pc Sr^{90}/g Ca would be equivalent to about 16,000 cases.

The above analogy assumes that Sr^{90} beta radiation induces leukemia of bone marrow origin at the same rate (per unit of absorbed

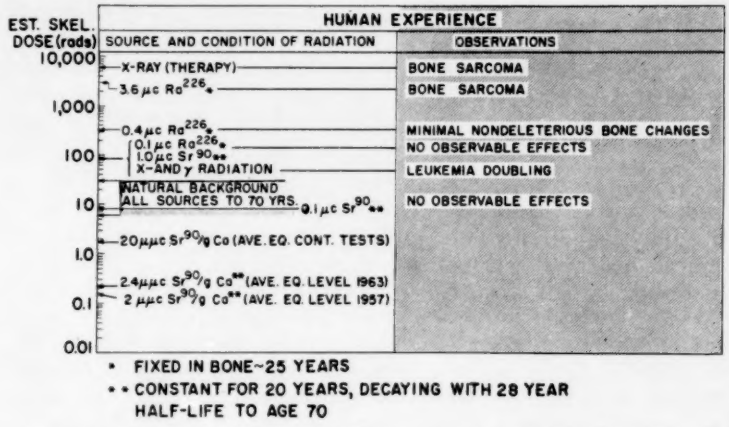


FIG. 8. Estimated Sr^{90} skeletal radiation dose in relation to human experience.

Table 9. Biological significance of present and predicted Sr^{90} *average maximum equilibrium levels and maximum permissible levels for occupational and nonoccupational exposure*

Sr^{90} level	MPL Nonocc. Exp. (100 pc/g)	MPL Occ. Exp. (1000 pc/g)	Min. bone changes	Min. sarcoma dose	Leukemia doubling dose
Present (~2 pc/g Ca)*	1/50	1/500	1/2000	1/20,000	1/500
Future (with continued testing)† (~20 ppc/g Ca)*	1/5	1/50	1/200	1/2000	1/40
100 ppc/g (MPL nonocc. exposure)	—	1/10	1/40	1/400	1/10
1000 ppc/g (MPL occup. exposure)	10	—	1/4	1/40	1

* Average maximum equilibrium level. A few persons may have five times the average.

† Assuming equilibrium with a continuous test rate of 10 MT equivalents of fission yield per year.

dose) as X- and γ-rays. Much of the beta radiation from Sr^{90} will be absorbed in the bone and not reach the hematopoietic tissues at all. Experiments by BRUES *et al.*[54] suggest that Sr^{89} (half-life 55 days, $E_\beta = 1.5$ MeV) administered to mice is relatively more spectacular as an osteosarcogenic agent than a leukemogenic agent. Furthermore, leukemia was not a significant finding in the radium dial painters[41,55] or in the radium-injection cases.[49]

Bone sarcoma is more apt to result from Sr^{90} than is leukemia. Human data on radiation-induced osteogenic sarcoma are not adequate to provide even the crudest estimate of the dose response relationship, the population doubling dose, or the fraction of normal population incidence (about 2/100,000) attributable to natural background.

Under the same conditions, the potential risk to the population from bone sarcoma, however, would be less than that calculated for leukemia, since its natural incidence in the population is lower than that of leukemia.

DISCUSSION

Past testing of nuclear weapons has produced between 5 and 6 Mc of Sr^{90} (equivalent to 50 to 60 MT of fission energy). About 90 per cent of the production has occurred since 1952 from testing of weapons in the megaton class. United States Pacific tests have been held under conditions that maximized local fallout (where it is of no concern because of the vast calcium reservoir of the ocean) and minimized world-wide contamination. Soil data suggest that about 1.6 Mc of Sr^{90} have been distributed as long-range fallout. The present stratospheric reservoir is estimated at about 2.4 Mc by LIBBY,[3] and at 1.1 ± 0.93 in this report. Present integrated surface deposition levels are such that the rate of Sr^{90} decay on the ground is almost equal to the rate of stratospheric fallout. If weapons tests were to stop, integrated surface deposition levels in the north temperature latitudes would probably increase by no more than 10 per cent, reaching a maximum in about 1963–1970.

Unfortunately, because of the locations of the U.S.A. and U.S.S.R. test sites and tropospheric and stratospheric meteorological phenomena, long-range fallout is maximized in the north temperate latitudes where over 80 per cent of the world's population lives. The present average soil level in the northern United States is about 35 mc/mile^2, and the average level elsewhere in the same general latitudes may be about 20 mc/mile^2. Deposition levels else where in the world are not potentially important with regard to general world health because of population distribution.

Estimates of average maximum Sr^{90} equilibrium bone levels for the northern United States and the north temperate population belt (from weapons tests to date) vary from about 1 to 4 pc/g Ca. Controversy over the issue of stopping or continuing bomb tests has resulted in greater apparent public confusion over the potential hazard of world-wide Sr^{90} fallout, than seems justified by the factor of 4 difference in estimates of average maximum equilibrium bone levels. This confusion has resulted largely from differences in choice of reference as to average maximum permissible Sr^{90} levels applicable to the general population, and differences in opinion as to an appropriate factor of allowance for nonhomogeneity of fallout and bone uptake.

LIBBY[2,3] and KULP,[56] before any authoritative statements regarding a Sr^{90} MPL for the general population had been issued, used the occupational MPL (1000 pc/g Ca) as a reference. Later the National and International Commissions for Radiological Protection recommended that the MPL for large segments of the general population should be one-tenth (100 pc/g Ca) that for occupational exposure. The U.S. National Academy of Sciences-National Research Council report[45] inferred that 50 pc/g Ca might be considered as a safe level for the general population. The British Medical Research Council report,[43] while acknowledging that the maximum allowable concentration of Sr^{90} in the bones of the general population should not be greater than 100 pc/g Ca, stated that immediate consideration should be required if the concentration in human bones showed signs of rising greatly beyond one-hundredth (10 pc/g Ca) of that corresponding to the maximum permissible occupational level. LAPP[57] has stated also that the MPL for the general population perhaps should be one-hundredth of

the occupational value. All of these numbers have been brought to public attention during the controversy over continued weapons tests.[6]

Confusion has been increased also by the use of various safety factors for nonhomogeneity of fallout and bone uptake. Articles have appeared in which no factor was used,[2,3,36] others have appeared in which factors of 5[9] and 10[57,58] were recommended.

The effect of choice of values for the acceptable general population MPL and the choice of safety factors for nonhomogeneity of distribution and uptake are shown by the data in Table 10. These data were derived by simple proportionality (maximum bone level from present tests: 50 MT; Acceptable MPL: X) and show the value in megatons of fission energy release (over a short period) required to bring the average maximum equilibrium bone levels of the population to the various permissible values that have been called to public attention. The table also indicates the effect of various nonhomogeneity safety factors on the world average population level. These data show a variation of a factor of about 1000 in the megaton equivalents of fission that could be detonated depending on whether one wishes to be ultraconservative and use the highest safety factor for nonuniformity and the lowest value recommended for the general population, or be the opposite and use the occupational MPL and no safety factor for nonuniformity. The most important point to the data is that they explain the principal reason for public confusion and show that the major areas of uncertainty are: (1) the maximum permissible level for Sr^{90} as applied to the general population, and (2) the deviation of equilibrium bone values from the mean.

The most important question regarding the

Table 10. *Allowable level (in megatons) of fission energy release as a function of various general population MPL's*

Source of equilibrium bone level estimate and region	Av. max. bone level (no more testing) (pc/g Ca)	Allowable MT of fission yield			
		10 pc/g Ca	50 pc/g Ca	100 pc/g Ca	1000 pc/g Ca
United States					
LIBBY[9]: ecological data	3.9–1.7	130–300	650–1500	1300–3000	13,000–30,000
KULP[36]: ecological data	2	250	1250	2500	25,000
KULP[24]: bone data	1.5	300	1500	3000	30,000
EISENBUD[11]: milk data	4	120	600	1200	12,000
This paper: ecological data	3.5	140	700	1400	14,000
This paper: bone data	1.9	250	1250	2500	25,000
North temperate latitudes					
This report: ecological data	3.6	140	700	1400	14,000
This report: bone data	1.9	250	1250	2500	25,000
World average					
KULP[36]: ecological data	1.3	380	1900	3800	38,000
This report: ecological data	3.1	160	800	1600	16,000
This report: bone data	1.7	150	1500	3000	30,000
World average (no factor for distribution)	2	250	1250	2500	25,000
Average $\times \frac{1}{5}$ (for nonuniformity)		50	250	500	5,000
Average $\times \frac{1}{10}$ (for nonuniformity)		25	125	250	2,500

potential hazard of long-range Sr^{90} fallout is in relation to future weapons testing. If there is an upper limit to the amount of Sr^{90} in the bones of the population that can be safely tolerated, then the megaton equivalents of fission products that can be contributed per year to the biosphere by all nations is limited.

If Sr^{90} contamination from weapons testing by all nations continues at the same rate as has occurred during the past five years (about 10 MT equivalents of fission yield per year), equilibrium will be reached in about 100 years. At equilibrium the amount of Sr^{90} which will disappear each year from the environment (due to radioactive decay) will equal the amount that is being produced and continuing weapons tests will not result in any further increase in the population bone levels.

LIBBY[9,14] and others [15] have predicted that soil and bone levels at equilibrium with a continued test rate equal to that of the past 5 years will be eight to thirteen times the present values. On the basis of present average maximum equilibrium Sr^{90} bone levels postulated from the considerations set forth in this paper, the bones of the United States population will reach a steady state with the present testing rate at a value of 17 to 31 pc/g Ca. The equilibrium value for the weighted average world population will be 15 to 28 pc/g Ca.

LIBBY[9] has stated that something between 5 and 20 pc/g Ca would be the average maximum Sr^{90} concentration in the bones of the United States population if testing continued indefinitely at the average rate of the past 5 years. If testing is continued at the same rate and pattern as in the past, KULP[24] predicted an equilibrium level will be approached in the North American population of about 8 ρc/g Ca in about 30 years, and NEWMAN[15] in testimony before the Congressional Subcommittee suggested equilibrium bone levels of about 90 ρc/cg Ca may be reached in the northern United States. The values given above show disagreement by a factor of about 10. If, however, we accept as a reasonable average the values developed in this paper, the average Sr^{90} radiation dose to the bones of the population of the northern United States, at equilibrium with continued testing at the past rate, may be about 20 to 30 per cent of the average radiation dose from natural background, or about 20 to 30 per cent of the maximum permissible level adopted by the National and International Commissions. Since individual variations may result in a small number of people accumulating Sr^{90} burdens that are five times the average, the radiation dose to these few individuals may approach as an upper limit 100 to 150 per cent of the recommended maximum level. If testing is continued at the present rate for 30 years, the average level of Sr^{90} in the population of the northern United States may be about 10 to 15 per cent of natural background. This may result in a few people approaching body burdens about 50 to 75 per cent of the recommended maximum. Sr^{90} burdens in the weighted world average population will be essentially the same.

These data suggest that the past rate of biospheric Sr^{90} contamination, if continued for several years, will not produce population bone levels that will exceed the maximum permissible levels accepted by the National and International Commissions. Although these predictions suggest little potential hazard from the present rate of world-wide Sr^{90} fallout, because of the great uncertainties involved, it is imperative that the problem be kept under constant scrutiny if weapons tests are to continue.

Present considerations show also that any and all nations cannot test high yield fission weapons indiscriminately without taking the risk of producing Sr^{90} bone levels in segments of the population that may exceed the MPL recommended by national and international agencies whose purpose is the protection of humanity from radiation damage.

REFERENCES

1. W. F. LIBBY, *Science* **123,** 567–660 (1956)
2. W. F. LIBBY, *Proc. Nat. Acad. Sci.* **42,** 365–390 (1956).
3. W. F. LIBBY, *Proc. Nat. Acad. Sci.* **42,** 945–962 (1956).
4. M. EISENBUD, *Global Distribution of Radioactivity from Nuclear Detonations, with Special Reference to Strontium-90.* Presented at the Washington Academy of Sciences meeting. Washington, D.C. (1956).
5. C. I. CAMPBELL, *Science* **124,** 894 (1956).

6. Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, First Session on the Nature of Radioactive Fallout and Its Effects on Man, Part 1, 27–29 May and 3 June, 1957, and Part 2, 4–7 June, 1957.
7. L. D. Marinelli, *Science* **124,** 122 (1956).
8. E. C. Anderson, R. L. Schuch, W. R. Fisher and W. H. Langham, *Science* **125,** 1273–1278 (1957).
9. W. F. Libby, *Radioactive Fallout.* Presented before the Spring Meeting of the American Physical Society. Washington, D. C. (1957).
10. L. Machta, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, First Session on the Nature of Radioactive Fallout and Its Effects on Man, Part 1, pp. 141–162. 27–29 May and 3 June, 1957.
11. M. Eisenbud, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, First Session on the Nature of Radioactive Fallout and Its Effects on Man, Part 1, pp. 554–591. 27–29 May and 3 June, 1957.
12. E. P. Hardy, Jr., *Health and Safety Laboratory Report.* New York Operations Office, HASL–1 (1957).
13. N. G. Stewart, R. N. Crooks and E. M. R. Fisher, British Atomic Energy Research Establishment (Harwell), AERE/HP/R 1701 (1955) and AERE/HP/R 2017 (1956).
14. W. F. Libby, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, First Session on The Nature of Radioactive Fallout and Its Effects on Man, Part 2, p. 1345. 4–7 June, 1957.
15. W. F. Newman, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, First Session on the Nature of Radioactive Fallout and Its Effects on Man, Part 2, pp. 1346–1348. 4–7 June, 1957.
16. W. F. Libby, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Eighty-Fifth Congress, First Session on The Nature of Radioactive Fallout and Its Effects on Man, Part 2, p. 1346. 4–7 June, 1957.
17. R. Menzel, *Science* **125,** 219–225 (1957).
18. K. Larson, UCLA–380 (1956).
19. H. I. Bowen and J. A. Dymond, *J. Exp. Bot.* **7,** 264–272 (1956).
20. G. V. Alexander, R. E. Nusbaum and N. S. MacDonald, *Analytical Data.* Presented at the Bio-Medical Program Directors meeting ay UCLA (1957.)
21. G. V. Alexander, R. E. Nusbaum and N. S. MacDonald, *J. Biol. Chem.* **218,** 911–919 (1956).
22. C. L. Comar, quoted in Committee Report *Deposition and Retention of Ingested Strontium 90 in the Skeleton.* Washington, D. C. (1957). Official use only.
23. H. Spencer, D. Laszlo and M. Brothers, *J. Clin. Invest.* **36,** 680–688 (1957).
24. W. R. Eckelmann, J. L. Kulp and A. R. Schulert, *Science* **127,** 266–274 (1958).
25. D. Laszlo, quoted in Committee Report *Deposition and Retention of Ingested Strontium 90 in the Skeleton.* Washington, D. C. (1957).
26. C. L. Comar, R. H. Wasserman and M. M. Nold, *Proc. Soc. Exp. Biol. Med.* **92,** 859–863 (1956).
27. R. M. Hodges, N. S. MacDonald, R. Nusbaum, R. Stearns, F. Ezmirlian, P. Spain and C. McArthur, *J. Biol. Chem.* **185,** 519–524 (1950).
28. K. K. Turekian and J. L. Kulp, *Science* **124,** 405–406 (1956).
29. E. M. Sowden and S. R. Stitch, *Trace Elements in Human Tissue. Part* 2. *Estimation of the Concentrations of Stable Strontium and Barium in Human Bone.* AERE/MRC/R 2030 (1956).
30. C. L. Comar, I. B. Whitney and F. W. Lengemann, *Proc. Soc. Exp. Biol. Med.* **88,** 232 (1955).
31. W. S. Woytinsky and E. S. Woytinsky, *World Population and Production, Trends and Outlook.* The Twentieth Century Fund, New York (1953).
32. 1955 *Year Book of Food and Agriculture Statistics.* United Nations-FAO, Rome (1956).
33. *Nutritional Charts* (11th Ed.). Research Dept. of H. J. Heinz Company, Pittsburgh, Pa. (1942).
34. W. H. Langham and E. C. Anderson, *Science* **126,** 205–206 (1957).
35. H. H. Mitchell, T. S. Hamilton, F. R. Steggerda and H. W. Bean, *J. Biol. Chem.* **158,** 625 (1945).
36. J. L. Kulp, W. R. Eckelmann and A. R. Schulert, *Science* **125,** No. 3241, 219–225 (1957).
37. R. J. Bryant, A. C. Chamberlain, A. Morgan and G. S. Spicer, *Radiostrontium Fallout in Biological Materials in Britain.* AERE/HP/R 2056, Harwell, Berks (1956).
38. R. F. Palmer and F. B. Queen, *Hanford Atomic Energy Works Report* HW–31242 (1956).
39. Recommendations of the Internal Commission on Radiological Protection. Supplement No. 6, *Brit. J. Radiol.* (1954).

40. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, National Bureau of Standards Handbook 52 (1953).
41. J. C. Aub, R. D. Evans, L. H. Hempelmann and H. S. Martland, *Medicine* **31,** 221–329 (1952).
42. *Permissible Dose from External Sources of Ionizing Radiation.* National Bureau of Standards Handbook 59 (1954).
43. *The Hazards to Man of Nuclear and Allied Radiations.* British Medical Research Council, Her Majesty's Stationery Office, London (1956).
44. *The Biological effects of Atomic Radiation.* Summary Reports. National Academy of Sciences–National Research Council (1956).
45. *Pathologic Effects of Atomic Radiation.* National Academy of Sciences–National Research Council, Publication 452,)1956).
46. E. B. Lewis, *Science* **125,** 965–972 (1957).
47. A. M. Brues, *Rad. Res.* **3,** 272–280 (1955).
48. A. M. Brues, *Commentary on the Modes of Radiation Injury.* International Conference on the Peaceful Uses of Atomic Energy (1955).
49. W. B. Looney, R. J. Hasterlik, A. M. Brues and E. Skirmont, *Amer. J. Roentgenol.* **73,** 1006–1037 (1955).
50. W. G. Cahan, H. Q. Woodward, N. L. Higinbotham and F. W. Stewart, *Cancer* **1,** 3–29 (1948).
51. W. M. Court Brown and J. D. Abbat, *Lancet* **268,** 1283–1285 (1955).
52. F. W. Spiers, *The Hazards to Man of Nuclear and Allied Radiations.* Her Majesty's Stationery Office, London (1956).
53. R. A. Dudley and R. D. Evans, *Radiation Dose to Man from Natural Sources.* Submitted to Joint Committee on Atomic Energy, Congress of the United States (1957).
54. A. M. Brues, *J. Clin. Invest.* **28,** 1286–1296 (1949).
55. W. B. Looney, *J. Bone Joint Surg.* **38**A, 175–218 (1956).
56. J. L. Kulp, W. R. Eckelmann and A. R. Schulert, *Science* **125,** 934 (1957).
57. R. E. Lapp, *Science* **125,** 933–934 (1957).
58. W. O. Caster, *Science* **125,** 1291 (1957).

Health Physics Pergamon Press 1958. Vol. 1, pp. 125–134

CURRENT STATUS OF THE INTERNAL DOSE PROBLEM*

K. Z. MORGAN
Director, Health Physics Division, Oak Ridge National Laboratory,
Operated by Union Carbide Nuclear Company, for U.S. Atomic Energy Commission,
Oak Ridge, Tennessee

Abstract—There are MPC values for many additional radionuclides in the revised handbooks (now ready to go to press) of the National Committee on Radiation Protection, NCRP, and of the International Commission on Radiological Protection, ICRP. The basic philosophy and principal standards used in obtaining these MPC values have not changed. The MPC values are calculated on the basis of a dose rate of 0.6 rem/week to the thyroid or skin, 0.56 rem/week to bone, 0.1 rem/week to the gonads or total body, and 0.3 rem/week to other organs of the body. In this revised edition the period of occupational exposure is 50 years (it was formerly 70 years). The GI tract is the critical body organ for more than 50 per cent of the 230 radionuclides listed in these publications. All MPC values are calculated on the assumption that the body burden of the radionuclides is an exponential function of time. In addition, in the few cases where sufficient data are available, calculations are made on the basis of a power function of time.

THE Handbooks on Maximum Permissible Internal Dose from Occupational Exposure of the National Committee on Radiation Protection (NCRP) and of the International Commission on Radiological Protection (ICRP) are now in the final stages of revision, and it is expected that the revised editions† will go to press in July 1958.

The present membership of the Internal Dose Committee of these organizations is as follows:

NCRP	ICRP
K. Z. MORGAN, Chairman	K. Z. MORGAN, U.S. Chairman
A. M. BRUES	W. BINKS, U.K.
P. W. DURBIN	A. M. BRUES, U.S.
G. FAILLA	W. H. LANGHAM, U.S.
J. W. HEALY	L. D. MARINELLI, U.S.
L. D. MARINELLI	W. G. MARLEY, U.K.
J. E. ROSE	G. J. NEARY, U.K.
SHIELDS WARREN	M. NAKAIDZUMI, Japan
J. B. HURSH	E. POCHIN, U.K.
	M. N. POBEDINKSI, U.S.S.R.
	C. G. STEWART, Canada

The basic philosophy and principal standards of reference used in the preparation of these handbooks have not changed. The principal changes are in details of application and the incorporation of more recent data.

The basic philosophy is that persons occupationally engaged in work with radioisotopes, nuclear reactors, X-ray equipment, and other accelerators capable of producing ionizing radiation must receive no unnecessary exposure but should be expected to receive on the average an accumulated RBE dose in excess of that permitted to the population at large. (See ICRU Handbook[1] 62 for definition of "RBE dose", rem, rad, etc.) We are often asked, "why do we have this discrimination or double standard?" The answer is that this is no exception to common practice. Each industry has its own unique hazards. The plumber, the welder, the housewife, the radiation worker, each has his or her particular brand of hazards. As shown by Fig. 1, our chance of survival as a people would be very low if each industry imposed a full measure of its hazards on the population at large. The maximum permissible integrated dose to occupational workers may be several hundred rems in a lifetime and can be expected to bring about certain biological changes and result in various types of damage but not in

* Presented 18 June 1957, at the Health Physics Society Annual Meeting, Pittsburgh, Pennsylvania.

† At the New York meeting of the I.C.R.P. in March 1958 it was recommended that the Internal Dose Handbooks of I.C.R.P. and N.C.R.P. should be combined into a single publication.

damage that is readily detectable or identifiable in the individual as definitely attributable to exposure to ionizing radiation. This occupational dose must not be sufficient to produce any observable acute damage, and the resulting chronic damage must not be sufficient for detection except from a statistical analysis of medical studies of many individuals following long periods of occupational exposure. The occupational damage permitted from ionizing radiation must not exceed that commonly accepted in other industries. For example, the increased incidence of disease, the number of cases of leukemia, the shortening of life span, etc., must not exceed that produced by other well-managed industries. A man working with ionizing radiation—the same as the chemist, the metallurgist, the housewife, and the painter—has a right to expect on the average to live a happy life and die of old age at 75.

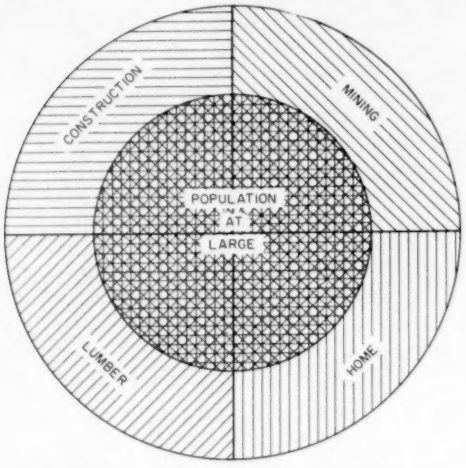


FIG. 1. The amount of shading is intended to symbolize the hazards. The center section shows the dose to the population at large, if each of the four professions were permitted to contribute its unique hazards to the environment.

The principal standards of reference in the Handbooks of the NCRP[2] and ICRP[3] for the selection and establishment of acceptable values of maximum permissible body burden, q, and of maximum permissible concentration, MPC, of the radionuclides in air, water, and food have been as follows:

(a) The values of MPC and the resultant q must be such that for continuous occupational exposure the RBE dose to the bone will not exceed that received from 0.1 μg of Ra^{226} and its resulting daughter products deposited in the bone.

(b) The values of MPC and the resultant q must be such that the weekly RBE dose to most critical body organs resulting from continuous occupational exposure will not exceed 0.3 rem* (bone, skin, thyroid, gonads and total body are exceptions to this rule).

A third standard has been added, viz.:

(c) The values of MPC and the resulting q must be such that the quarterly RBE dose to the gonads or to the total body will not exceed 0.1 rem, and the average dose to age N will not exceed $5(N-18)$ rem.

The principal change in (a) above in the 1958 revision of the NCRP and ICRP handbooks is to assume that 30 per cent of the daughter products of Ra^{226}, rather than 55 per cent, are retained[4] in the body. Thus, the weighted energy of Ra^{226} and its daughter products becomes 110 MeV instead of 162 MeV, so that the equation for maximum permissible body burden of bone seeking radionuclides is now:

$$q = \frac{0.1\ (0.99)}{f_2} \times \frac{110}{\Sigma E(\mathrm{RBE})N} = \frac{11}{f_2\ \Sigma E(\mathrm{RBE})N} \qquad (1)$$

in which

f_2 = fraction of radionuclide in bone of that in total body
E = energy (MeV) absorbed in bone per disintegration of the radionuclide
(RBE) = relative biological effectiveness of radiation
N = relative damage factor
0.99 = value of f_2 for Ra^{226}
0.1 = value of q for Ra^{226}

This change in equation (1) means that 0.1 μg of Ra^{226} plus 30 per cent of its daughter products corresponds to a weekly RBE dose to the bone of 0.56 rem rather than 0.82 rem. Both values are for $(\mathrm{RBE})_\alpha = 10$. Thus equation (1) gives

* This dose rate may now (August 1958) be averaged over a 13 week period.

values that are only about twice (rather than three times) those specified by (b) above. A very minor change in the application of equation (1) has been that it is now applied to all α, β and other particulate emitting radionuclides (rather than just α-emitting radionuclides) for which bone is the critical body organ. This extension of the radium comparison to all bone seeking radionuclides except those that emit only X- or γ-radiation seems reasonable and has been implied in many recent discussions but has not been specifically included in the past publication of the handbooks, although the values of q and MPC for Sr^{89} and Sr^{90} were based on a radium comparison made at the Chalk River Conference in 1949.

There are very few changes in (b) above in the 1958 revision of the NCRP and ICRP handbooks. The maximum weekly RBE dose for the thyroid was set at 0.6 rem in the ICRP handbook, and this value is now used in the NCRP handbook. Also, the maximum weekly RBE dose is set at 0.6 rem in the revisions of both handbooks when skin is the critical body organ. The cases in the new publications in which 0.6 rem is the limiting weekly RBE dose are summarized in Table 1.

Table 1. *Cases in which the limiting weekly dose RBE is* 0.6 *rem*

Skin critical tissue for		Thyroid critical tissue for	
$_{14}Si$	(1st)	$_{53}I$	(1st)
$_{16}S$	(1st)	$_{75}Re$	(1st)
$_{75}Re$	(2nd)	$_{85}At$	(1st)
		$_{50}Sn$	(2nd)

The notations in parentheses indicate in which cases the skin or thyroid is the first or second critical body organ.

In the past editions of the Internal Dose Handbooks of the NCRP and ICRP, the period of occupational exposure, t, was taken as 70 years, and as a result the term $[1 - \exp(-0.693\, t/T)] \approx 1$ or equilibrium was reached in all cases with the exception of Th^{232}, Pu^{239}, Sm^{151}, and Ra^{226}. In the new editions of the 1958 handbooks the more realistic period of occupational exposure of 50 years is used, and in this case additional radionuclides have effective half-lives, T, sufficiently long that equilibrium is not reached in a period $t = 50$ years. The cases in the new publications in which equilibrium is not reached in a period of occupational exposure of 50 years are given in Table 2.

From Table 2 it can be seen that the reduction of the work period from 70 to 50 years has resulted only in a minor increase in the MPC values for some of the radionuclides (i.e. a maximum increase of 27 per cent). The fact that there are now nineteen radionuclides (rather than four) that do not reach equilibrium is the result of new data for the effective half-life T and the inclusion of many additional radionuclides in the revised Internal Dose Handbooks.

Often there are questions of why the MPC values in the NCRP and ICRP Internal Dose Handbooks have been based on continuous exposure (24 hr/day and 7 days/week) rather than on a 40 hr week. The only answers to this question are that there is considerable variation in the number of hours worked per week, and in some cases it is convenient to use the handbook values as a guide in estimating MPC values for continuous, non-occupational exposure. The MPC values in the revised ICRP and NCRP handbooks will be given for continuous exposure and for a period of 40 hr per week in conformance with what has become a standard work practice in many countries. This change will eliminate the need for multiplying the MPC values in the handbooks by a factor of 3* in order to make them applicable to the common occupational exposure.

As mentioned above, the MPC values in the Internal Dose Handbooks sometimes serve as a guide in estimating MPC values for continuous, non-occupational exposure. For example, values

* This factor of three results from the fact that the air and water intake in an 8 hr work period is about half that in the 24 hr day and the common work period in the U.S. is 8 hr/day, 5 days/wk, and 50 wk/yr, (i.e. $2 \times 7/5 \times 52/50 = 3$). The multiplication factors that should be applied ad interim to the occupational exposure MPC values to make them applicable to non-occupational exposure are: 1) 1/30 for application in the vicinity of the control area and 2) 1/100 for application to the population-at-large.

Table 2. Radionuclides which do not reach equilibrium in the body in exposure period

Z	Radionuclide	T_r (yr)	T_b (yr)	T (yr)	% equil. reached in 50 years	% equil. reached in 70 years
38	Sr^{90}	28	200	24.6	76	86
88	Ra^{226}	1622	43.8	43.8	56	67
89	Ac^{227}	21.8	200	19.7	83	91
90	Th^{230}	8.0×10^4	200	200	16	22
90	Th^{232}	1.39×10^{10}	200	200	16	22
93	Np^{237}	2.20×10^6	200	200	16	22
94	Pu^{238}	89.6	200	61.9	43	54
94	Pu^{239}	2.44×10^4	200	198	16	22
94	Pu^{240}	6.6×10^3	200	194	16	22
94	Pu^{241}	13.2	200	12.4	94	98
94	Pu^{242}	3.8×10^5	200	200	16	22
95	Am^{241}	462	200	140	22	29
95	Am^{243}	8×10^3	200	195	16	22
96	Cm^{243}	35	200	29.8	69	80
96	Cm^{244}	18.4	200	16.8	87	94
96	Cm^{245}	2×10^4	200	198	16	22
96	Cm^{246}	6.6×10^3	200	194	16	22
98	Cf^{249}	4.7×10^2	200	140	22	29
98	Cf^{250}	10	200	9.52	97	99

of one-tenth or one-hundredth of the occupational esposure values have been suggested and used by various organizations as interim MPC values for non-occupational exposure. Such values must be used, however, with considerable caution because, for example, the MPC values for occupational exposure to Cs^{137} are based on a limited weekly RBE dose of 0.3 rem to muscle, and one-hundredth of this value may have little or no relationship to a limit "the order of natural background", as set by the ICRP† for genetic exposure of the population-at-large to all man-made sources of ionizing radiation. At present we have very little data on the gonad exposure from the various radionuclides that may be taken into the body by food, water, and air. A limited number of such studies are underway, but until such data are more complete, MPC values for the population-at-large must be of a tentative nature.

The basic reference (c) above is a new limitation designed to reduce the accumulated occupational RBE dose. For the past decade it has been apparent to the health physicists that although the 0.3 rem weekly RBE dose is satisfactory as a day-to-day operating level, it would not be satisfactory to integrate this rate over a large fraction of the occupational life. It is commendable that all health physicists have recognized this fact, and motivated by their understanding of the problem, and guided by a large measure of good judgment have insisted upon and enforced their own unofficial factors of safety. At times it was not too easy for the health physicists to convince other scientists, engineers, operators, and management that these unofficial factors of safety must be complied with. Now that the permissible integrated RBE dose for external exposure and for internal

† The report of ICRP presented to the Ninth International Congress of Radiology in Mexico in 1956 states, "Until general agreement is reached, it is prudent to limit the dose of radiation received by gametes from all sources additional to the natural background to an amount of the order of the natural background in presently inhabited regions of the earth".

Table 3. Maximum permissible external dose

Age interval (years)	NCRP Values				ICRP values			
	Old HB 59 values		New values		Old values		New values	
	Rate (rems/ week)	Dose (rems)	Rate‡ (rems/ week)	Dose (rems)	Rate (rems/ week)	Dose (rems)	Rate‡ (rems/ week)	Dose (rems)
−0.75 – 18*	0.03	29	0.01	10*	0.03	29	0.0025	2*
18 – 45	0.3	423	0.1	141	0.3	423	0.1	141
45 – 70	0.6	783	0.1	130	0.3	391	0.1	130
Total Dose to Age 70		1235†		281		843†		273

* Preoccupational exposure period. Whether this RBE dose to age 18 is 10 rems or 2 rems depends upon whether it is assumed the person lives in the "vicinity of the control area" or is an average individual of the population-at-large.

† Although the NCRP and ICRP regulations permitted the accumulation of RBE doses of 1235 rem and 843 rem, respectively, they recommended avoiding all unnecessary exposure and as explained above the health physicist applied large safety factors so that in practice occupational exposures were much less than these values.

‡ These values may be averaged over 13 weeks or increased by 12/5 if the age limitation $5(N - 18)$ is not exceeded.

exposure where the total body or the gonads are the critical body organs has been reduced officially, the problems of the health physicist are very much simplified. Also, the NCRP handbook permitted a doubling of the RBE dose rate at age 45. This rule was objected to by most health physicists and was not applied by them because it was tantamount to making an announcement to the faithful, long-time employees, who had accumulated the most exposure, that they would be permitted to double their exposure rate as a special "service award". This provision for doubling the external exposure rate at age 45 will be dropped from the revised NCRP handbooks. The manner in which the RBE dose may be accumulated by an occupational worker in compliance with the NCRP and ICRP regulations is given in Table 3.

From Table 3 it is to be observed that the new value of permissible accumulated RBE dose for occupational exposure to age 70 is about the same for the NCRP as for the ICRP (i.e. 281 rem permitted by NCRP and 273 rem permitted by ICRP) and that there has been a reduction by a factor of 4.4 in the NCRP value and of 3 in the ICRP value. It should be stressed that these new limits in Table 3 will not be applied to internal dose problems except where total body or gonads are the critical body organs. So far no data are available that indicate the gonads are the critical body organs for occupational exposure for any of the radionuclides. However, the total body is the critical body organ for the radionuclides listed in Table 4.

For the previously mentioned radionuclides the maximum permissible weekly exposure is 0.1 rem. It is to be noted that noble gases are included as one of the groups in this list. This results from the fact that the MPC values in this case are based on a submersion calculation, viz., the RBE dose is that delivered externally by a cloud of noble gas formed into an infinite semisphere* about the person. The 0.1 rem weekly RBE dose limitation should be applied

* Two changes have been made in the case of MPC calculations for noble gases: (1) the weekly RBE dose is limited to 0.1 rem rather than 0.3 rem, and (2) the RBE dose is assumed to be that delivered from half the solid angle rather than the entire solid angle. The result is that the MPC values have been reduced by a factor of two-thirds.

Table 4. Total body as the first critical organ

Based on element distribution	Based on submersion values		Based on dose distribution		
$_{1}H^{3}$	$_{18}A^{37}$		$_{4}Be^{7}$	$_{33}As^{74}$	$_{55}Cs^{131}$
$_{11}Na^{24}$	$_{18}A^{41}$		$_{15}P^{32}$	$_{33}As^{76}$	$_{55}Cs^{134m}$
$_{17}Cl^{36}$	$_{36}Kr^{85m} + Kr^{85}$	19%	$_{16}S^{35}$	$_{33}As^{77}$	$_{55}Cs^{134}$
$_{17}Cl^{38}$	$_{36}Kr^{85m}$	81%	$_{19}K^{42}$	$_{37}Rb^{86}$	$_{55}Cs^{136}$
$_{35}Br^{82}$	$_{36}Kr^{85}$		$_{27}Co^{57}$	$_{40}Zr^{95}$	$_{55}Cs^{137}$
$_{11}Na^{22}$	$_{36}Kr^{87}$		$_{27}Co^{58m}$	$_{41}Nb^{95}$	$_{56}Ba^{131}$
	$_{54}Xe^{131}$		$_{27}Co^{58}$	$_{43}Tc^{99m} + Tc^{99}$	$_{74}W^{187}$
	$_{54}Xe^{133}$		$_{27}Co^{60}$	$_{47}Ag^{105}$	$_{75}Re^{183}$
	$_{54}Xe^{135} + Cs^{135m}$		$_{30}Zn^{65}$	$_{47}Ag^{110m}$	$_{79}Au^{196}$
			$_{33}As^{73}$	$_{51}Sb^{122}$	

to all MPC calculations for noble gases or any other relatively inert gases if they emit γ-radiation or β-rays with sufficient energy to penetrate the protective epidermal layer and result in essentially a total body exposure problem.

The MPC values for H_2^3 and $C^{14}O_2$ are not determined on the basis of a submersion calculation but rather on the basis of a lung exposure because of their low energy β-radiation (i.e. $E_\beta = 0.0189$ MeV for H^3 and 0.155 MeV for C^{14}). On the other hand, if a MPC value were obtained for the 10.1 min N_2^{13} which emits a β^+-ray with an energy of 1.24 MeV, the calculation should be based on a submersion calculation.

The group of radionuclides listed in Table 4 under the "Element distribution" comprises radioisotopes of elements that are so uniformly distributed throughout the body that the RBE dose is almost uniform and presents essentially a problem of total body exposure. The group listed under "Dose distribution" is a large fraction of the total (i.e. 29 out of a total 44) and includes all the radionuclides that deliver a weekly RBE dose between 0.1 and 0.3 rem to some specific organ (such as bone, liver, kidney, spleen, etc.) while delivering a weekly RBE dose of 0.1 rem to the total body. In other words, in these cases the total body would not be the critical body organ except for the fact that the weekly RBE dose permitted to the total body (0.1 rem) is less than the weekly RBE dose (0.3 to 0.6 rem) permitted to other body organs.

In cases in which the critical body organ is not the total body, gonad, skin, thyroid, or bone (e.g. the critical body organ is the lung, GI tract, liver, kidney, spleen, muscle, etc.) the maximum permissible weekly RBE dose is 0.3 rem. Table 5 summarizes the values of weekly RBE dose and accumulated dose to age 70 that were assumed in making the calculations of the MPC values for occupational exposure given in the new NCRP and ICRP Internal Dose publication.

Table 5. Summary of RBE dose rates and integrated RBE dose to critical body organs to age 70 (As permitted by the NCRP and ICRP Occupational Exposure Values in the New Internal Dose Publication)

Critical body organ	Weekly dose* (rem)	Integrated† dose (rem)
Bone	0.56	1520
Skin and thyroids	0.6	1630
Total body and gonads	0.1	271
All other organs‡	0.3	815

* May be averaged over 13 weeks.

† In order to obtain the total permitted dose to age 70, the permitted non-occupational dose to age 18 must be added to these values.

‡ Lungs, muscle, spleen, kidneys, liver, GI tract, etc.

Some of the values in Table 5 appear to be quite high. Perhaps they are, and if so, this should be an added incentive to maintain the total body burden q of the radionuclides at as low a value as possible. However, it must be remembered that all the higher values in Table 5 are for partial body exposure (not total body exposure). At the present time we do not know whether 1520 rem to the bone presents more or less of a hazard than 270 rem to the total body. In fact, at the present time there is some indication, but no proof, that damage to humans at these low rates is linearly proportional to the accumulated dose. There is some evidence that permits us to speculate that 270 rems to the total body may shorten the average life span[5–11] by about 2 years, increase the incidence of leukemia[12–17] by a factor of 5, and increase the number of spontaneous mutations[18] in the offspring by a factor of about 5. Perhaps these hazards are of the same order of magnitude as those to which the painter, the arc welder, the taxi driver, and other occupational groups are subjected. In any case, very few occupations are devoid of serious hazards, and every effort and constant vigilance must be maintained to minimize these hazards. Fortunately, to the present time few employees have reached the newly proposed exposure rates listed in Table 5. In fact, the total exposures (external and internal) at our Laboratory (Oak Ridge National Laboratory) have been less than 5 per cent of the new limits for our average employee and less than 50 per cent of the new limits for the average exposure of the ten employees with the highest exposure records excluding our recent case of an over-exposure.*

* This over-exposure occurred on 4 October 1957 when an employee mistakenly unlocked a cell door and entered a forbidden area used for storage of waste solutions. During a period of about 1 min until he realized his mistake and hurried from the cell he received a total body exposure of 63 rad. Our medical department has been unable to detect any indication of biological damage to this individual, but in order to limit the RBE dose to age N to that given by the equation $5(N-18)$ rem, this person (age 22) will be limited to essentially radiation free work for the next 9 or 10 years. Although this equation does not appear in the past regulations, it will be a part of the 1958 revision of the NCRP Handbook 59 and we are already applying it to our work at ORNL.

One addition of considerable interest in the recent revisions of the Internal Dose Handbooks is the inclusion of a discussion of the application of the power function in obtaining q and values of MPC. For example, in the simplest cases (where $T_r \gg T_b$) the body burden q after an amount a has been ingested per day for L days may be given by:

$$q = Aaf_1 \int_0^L (L - \tau)^{-n}\, dt = \frac{Aaf_1}{1 - n} L^{1-n} \quad (2)$$

in which $a = 2200\ (\mathrm{MPC})_w$ for ingestion and $2 \times 10^7\ (\mathrm{MPC})_a$ for inhalation

f_1 = fraction going from GI tract to blood

A and n are constants.

Only limited data of sufficient extent are available for the application of the power function. A summary of the principal cases in which the power function is applied and some provisional data are given in Table 6.

One of the big differences between the past Internal Dose Handbooks of NCRP[2] and ICRP[3] has been that MPC values based on the GI tract were given in the ICRP handbook but not in the NCRP handbook of earlier publication date. This difference came about due to the fact that at the time of publication of the NCRP handbook on Internal Dose, it was felt that the tissue of the GI tract was less sensitive to radiation damage than other body tissue. Later studies did not support this view and so when the Internal Dose Handbook of ICRP was published, the MPC values based on the GI tract as the critical body tissue were included along with the MPC values based on other body organs as the critical body tissue. It turned out that in the ICRP Internal Dose Handbook 71 per cent of the MPC values for ingestion and 41 per cent of the MPC values for inhalation were based on the GI tract as the critical body organ. Therefore, in the revisions of both handbooks MPC values based on the GI tract as the critical body organ are given for each radionuclide. It is of interest that in the revised editions of the handbooks the GI tract is the critical body organ for 74 per cent of the cases of ingestion and for 33 per cent of the cases for inhalation. Another reason for including in the revised handbooks MPC values based on the GI tract and on the total

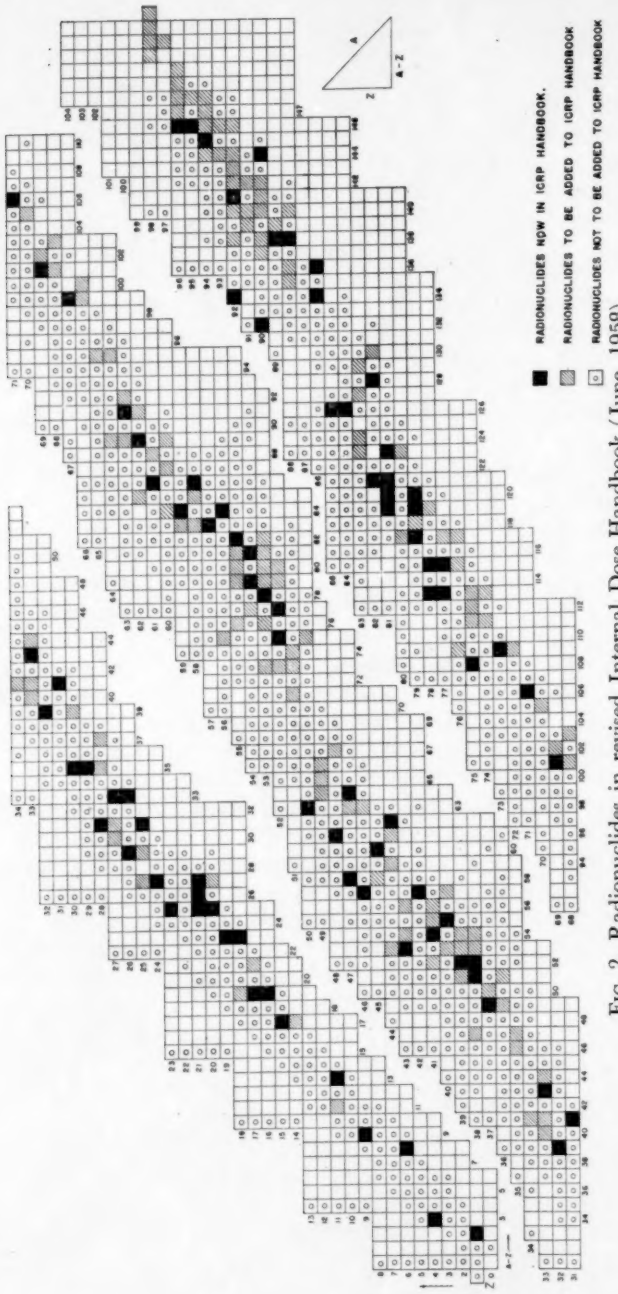


FIG. 2. Radionuclides in revised Internal Dose Handbook (June, 1958).

Table 6. Radionuclides for which the power function is applied

Radionuclide	Constant A	Constant n	$(MPC)_w$	$(MPC)_a$
Ra^{226}	0.54	0.52	1.2×10^{-6}	1.0×10^{-10}
Pu^{239}	0.99	0.01	4.0×10^{-5}	5.3×10^{-13}
U–Nat	0.60	0.50	8.4×10^{-4}	3.7×10^{-11}
Sr^{90}	0.65	0.35	6.5×10^{-6}	5.5×10^{-10}

body for all the radionuclides—regardless of whether they represent the first critical body organ—is that in many cases they are the second critical body organ but become the first critical body organ for a mixture of the same radionuclides.

Undoubtedly the most important change in the new Internal Dose Handbook is the inclusion of MPC values for many additional radionuclides and for the insoluble as well as the soluble chemical forms. Values were given for seventy radionuclides in Handbook 52[2] of the NCRP and for 86 radionuclides in the old ICRP Handbook.[3] The 1958 revisions of both the NCRP and the ICRP handbooks will contain MPC values for 233 radionuclides—an increase of over 300 per cent. The magnitude of the present revision of the Internal Dose Handbooks (where in general no radionuclides with half lives less than 1 hr were considered) and the extent of future work to be done is indicated in Fig. 2.

The present 1958 revisions of the two Internal Dose Handbooks have required about 5 man-years of hard labor.* Much of this type of work will continue indefinitely, and we have already begun to compile data for future revisions of the Internal Dose Handbooks. Below is a list of some of the new problems under consideration for future revisions.

(1) Selection of MPC values for single exposure.

(2) Selection of MPC values for human tracer studies.

(3) Selection of MPC values based on gonad exposure.

(4) Consider influence of secondary factors.

(5) Chronic biological studies of radionuclides.

Hundreds of changes are made in the terms used for the individual radionuclides in the calculation of the q and MPC values in the revised 1958 Internal Dose Handbooks. Such changes can be expected in the future in each revision of the handbooks as more physical, biological, and medical data become available. In conclusion, it is apparent that many of the q and MPC values do not contain the desired accuracy and are subject to continual refinement. However, in general, the values given probably represent the best data now available.

* The revisions of the Internal Dose Handbooks represent primarily the work of M. J. Cook, J. M. Mitchell, M. R. Ford, J. R. Muir, W. S. Snyder, M. G. Wright, and K. Z. Morgan. We have been assisted through constant contacts with the members of the Internal Dose Committee.

REFERENCES

1. *Report of the International Commission on Radiological Units and Measurements*, Handbook H–62 National Bureau of Standards. Washington, D.C. (1957).
2. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, National Committee on Radiation Protection, National Bureau of Standards, Handbook H–52. Washington, D.C. (1953).
3. Recommendations of the International Commission on Radiological Protection *Brit. J. Radiol.* Supplement No. 6 (1955).
4. W. P. Norris, T. W. Speckman and P. E. Gustafson, *Amer. J. Roentgenol.* **73,** 785–802 (1955).
5. R. D. Boche, *Observations on Populations of Animals Exposed to Chronic Roentgen Radiation.* Report MDDC–204 (1947).
6. R. D. Boche, *Effects of Chronic Exposure to X Radiation on Growth and Survival.* National Nuclear

Science Series, Div. 6, Chap. 10, Vol. 2. McGraw-Hill, New York (1954).

7. H. A. Blair, *A Formulation of the Injury, Life Span, Dose Relations for Ionizing Radiations. I. Application to the Mouse.* Unclassified University of Rochester Report, UR–206 (1952).
8. H. A. Blair, *A Formulation of the Injury, Life Span, Dose Relations for Ionizing Radiations. II. Application to the Guinea Pig, Rat and Dog.* Unclassified University of Rochester Report, UR–207 (1952).
9. H. A. Blair, *Shortening of Life Span by Ingested Radium, Polonium, and Plutonium.* Unclassified University of Rochester Report, UR–274 (1953).
10. H. A. Blair, *Recovery from Radiation Injury in Mice and Its Effects on LD-50 for Durations of Exposure Up to Several Weeks.* Unclassified University of Rochester Report, UR–312 (1954).
11. H. Jones, *Aging effects of Radiation.* Proceedings of the Health Physics Society meeting at University of Michigan, Ann Arbor, Mich. (1956).
12. P. S. Henshaw and O. W. Hawkins, *J. Nat. Cancer Inst.* **4,** No. 4, 339–346 (1944).
13. H. C. March, *Amer. J. Med. Sci.* **220,** No. 3, 282–286 (1950).
14. S. Peller and S. Pick, *Amer. J. Med. Sci.* **224,** No. 2, 154–158 (1952).
15. H. C. March, *Radiology* **43,** No. 3, 275–278 (1944).
16. S. Warren, *J. Amer. Med. Assoc.* **62,** No. 5, 464–468 (1956).
17. W. M. Court-Brown and R. Doll, *Leukaemia and Aplastic Anaemia in Patients Irradiated for Ankylosing Spondylitis.* Medical Research Council, Special Report Series, no. 295 (1957).
18. H. H. Plough, *Nucleonics* **10,** No. 8, 16–20 (1952).

Health Physics Pergamon Press 1958. Vol. 1, pp. 135–140

AEROSOL ACTIVITY FROM EXPERIMENTAL LOW POWER REACTOR OPERATION

P. R. BOLTON, E. C. BARNES, E. D. DURKOSH and W. D. SMALL

Industrial Hygiene Department, Bettis Plant,*
Westinghouse Electric Corporation, Pittsburgh, Pa.

(*Received* 9 *December* 1957)

Abstract—The radioactive aerosol resulting from the low power critical operation of light water moderated experimental reactors has been studied. This phenomenon has been found in conjunction with plastic and unclad fuel elements. The activity found is attributed to several of the decay daughters of the noble gas fission products. It is characterized by an effective half-life of about 25 min and is interpreted to be of low toxicity. The authors suggest the use of exhaust ventilation or respirators when entering the critical assembly rooms after certain operations to keep the exposure to a minimum.

DURING experiments with a low neutron flux, heterogeneous, light water moderated and cooled critical assembly, a beta–gamma radioactive aerosol was detected in the assembly room under certain conditions of operation. A study was made to identify the aerosol and to evaluate and control the exposure from it.

In an experimental critical assembly, the core make-up and the operating conditions can be altered in various ways. For the purposes of this paper, the most significant variation in the experiments was in the type of fuel elements employed. The earlier experiments used metal clad enriched uranium alloy fuel elements. Later, fuel elements of unclad metal alloy, plastic clad alloy and plastic tapes impregnated with uranium were used. The plastic material was a crystalline polyethylene compound extruded to completely envelop the fuel elements or it was mixed with uranium and extruded as an impregnated tape.

The core assembly is supported inside an open-top cylindrical tank or pot. The tank is supported by a stand which also serves as a work platform. The pot and stand are housed in a concrete shielded room adjacent to the control room. Access to the assembly room is through a sliding shield door which is interlocked with a safety scram circuit in the critical assembly control circuit. Shut-down of the reactor is effected by the insertion of control rods into the core and by dumping the water moderator from the core pot. For certain experiments it is necessary to enter the assembly room as soon as shut-down occurs to remove irradiated foils or to change mechanical components.

The existence of a problem was first recognized when significant radioactive contamination was detected on personnel leaving the assembly room by the radiation monitoring instruments located at the door. Up to this time, no contamination had occurred during months of operation. It was thought that this contamination might have come from handling a component which had faulty cladding. A search for the source of the contamination revealed that the assembly room in general was significantly contaminated immediately after shut-down. No surface contamination was detected 12 hr later even though no effort had been made to decontaminate the room. Obviously, the activity was short-lived and was directly related to certain operating conditions. It was observed that the contamination occurred only when unclad, plastic clad or plastic impregnated fuel elements were used and had not occurred when metal clad fuel elements were used.

An air sample quickly established that the problem

* Bettis Plant is operated for the U.S. Atomic Energy Commission by Westinghouse Electric Corporation.

was one of airborne activity rather than physical transfer of surface contamination by personnel working in the room. As a precaution, the use of supplied air respirators was initiated immediately until the problem could be quantitated.

Preliminary air samples were collected on Whatman 41 filter paper and counted in both an end-window Geiger Counter and an internal proportional counter to determine the half-life of the material. Fig. 1 shows that the collected samples had a half-life of about 20 to 24 min for the first 1–1½ hr after collection, and it then appeared to increase to 25–30 min for the next 1–2 hr. Of the seventy-five or more short-term air samples eventually collected, the initial half-life ranged from 18 to 33 min

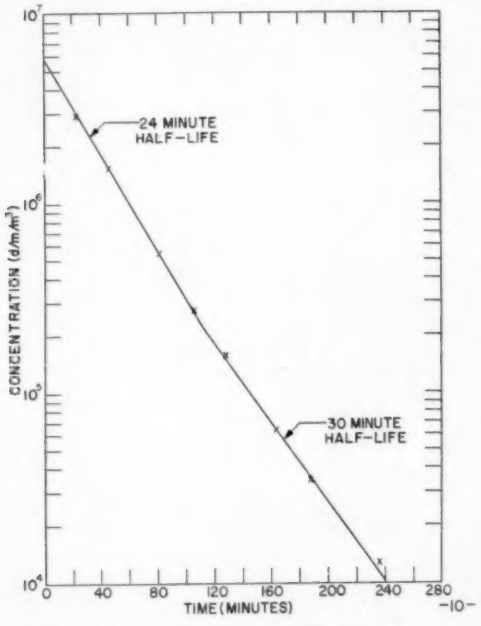


FIG. 1. Decay curve of aerosol activity.

with 85 per cent having a half-life of less than 25 min. There appeared to be no correlation between the half-life of the radioactive aerosol and the operating conditions of the critical assembly (i.e. neutron flux level, length of run, etc.) or between the half-life and the time of sample collection after shut-down of the critical assembly.

A comparison of some sampling methods was made. Simultaneous samples were collected using Millipore filters, Whatman 41 filters and three Greensburg-Smith impingers containing three different solutions. The Whatman and Millipore filters collected about five times as much activity per unit volume of air sampled as the impinger. More than 99 per cent of the total activity collected by two Millipore filters in series was collected by the first Millipore. Subsequent tests on Whatman 41 filter paper indicated that it had a collection efficiency of approximately 90 per cent when compared to the Millipore filter. Further air sampling was done using only Whatman 41 and Millipore filters. The lower level of detection with these filters was approximately 1000 d/min per m^3 (5×10^{-10} $\mu c/cm^3$).

The plastic clad and plastic impregnated fuel elements had been indicated as a possible source of the aerosol. The beta–gamma activity suggested that fission products were involved, and it appeared that the important ones would be those having the following characteristics:

(1) Have a high yield.

(2) Diffuse through semi-permeable solids to become airborne.

(3) Have a half-life in the 10–30 min range.

(4) Be soluble in water.

The two maxima of the fission product yield curve have mass numbers 95 and 139, which correspond with isotopes in the bromine, krypton, rubidium and iodine, xenon, cesium ranges respectively. The high yields (6.4 per cent) of these isotopes indicate that they would have a relatively large growth during the short periods of operation of the critical assemblies.

Of the elements previously mentioned, a number of them are metals and their rates of diffusion in solids are low. Rubidium and cesium are metallic and would remain in the fuel elements nearly at the point of origin. Any recoil atoms of rubidium and cesium which might escape from the fuel during operation would react chemically with the moderating water and not have the opportunity to get into the air. Likewise, the decay products of rubidium and cesium (strontium and barium) which might be formed in the water will remain there because of chemical reaction.

The bromine and iodine fission products are liquids at the temperatures in question and will, for the most part, be contained in the fuel element near the point of formation. Those that escape the fuel element as a result of recoil will remain dissolved in the moderating water. As evidence that no iodine and bromine were present, there was no change in the counting rate (when corrected for decay) of a Millipore filter sample after it was heated.

The isotopes of krypton and xenon formed in the fission process, being noble gases, have some freedom of motion in the fuel elements. Those atoms of

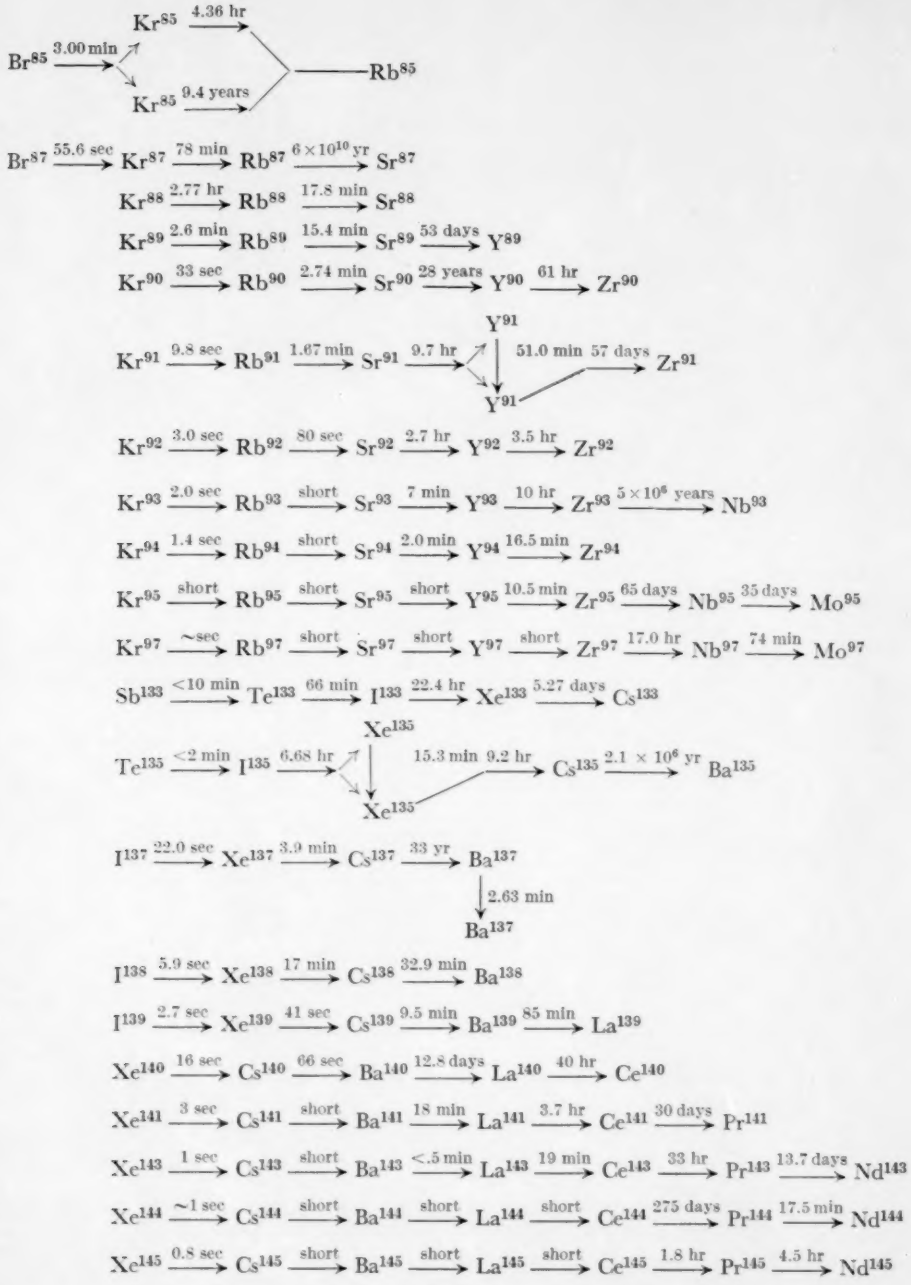

Br85 3.00 min Kr85 4.36 hr Kr85 9.4 years Rb85
Br87 55.6 sec Kr87 78 min Rb87 6×10^10 yr Sr87
Kr88 2.77 hr Rb88 17.8 min Sr88
Kr89 2.6 min Rb89 15.4 min Sr89 53 days Y89
Kr90 33 sec Rb90 2.74 min Sr90 28 years Y90 61 hr Zr90
Kr91 9.8 sec Rb91 1.67 min Sr91 9.7 hr Y91 51.0 min Y91 57 days Zr91
Kr92 3.0 sec Rb92 80 sec Sr92 2.7 hr Y92 3.5 hr Zr92
Kr93 2.0 sec Rb93 short Sr93 7 min Y93 10 hr Zr93 5×10^6 years Nb93
Kr94 1.4 sec Rb94 short Sr94 2.0 min Y94 16.5 min Zr94
Kr95 short Rb95 short Sr95 short Y95 10.5 min Zr95 65 days Nb95 35 days Mo95
Kr97 ~sec Rb97 short Sr97 short Y97 short Zr97 17.0 hr Nb97 74 min Mo97
Sb133 <10 min Te133 66 min I133 22.4 hr Xe133 5.27 days Cs133
Te135 <2 min I135 6.68 hr Xe135 Xe135 15.3 min 9.2 hr Cs135 2.1 × 10^6 yr Ba135
I137 22.0 sec Xe137 3.9 min Cs137 33 yr Ba137 2.63 min Ba137
I138 5.9 sec Xe138 17 min Cs138 32.9 min Ba138
I139 2.7 sec Xe139 41 sec Cs139 9.5 min Ba139 85 min La139
Xe140 16 sec Cs140 66 sec Ba140 12.8 days La140 40 hr Ce140
Xe141 3 sec Cs141 short Ba141 18 min La141 3.7 hr Ce141 30 days Pr141
Xe143 1 sec Cs143 short Ba143 <.5 min La143 19 min Ce143 33 hr Pr143 13.7 days Nd143
Xe144 ~1 sec Cs144 short Ba144 short La144 short Ce144 275 days Pr144 17.5 min Nd144
Xe145 0.8 sec Cs145 short Ba145 short La145 short Ce145 1.8 hr Pr145 4.5 hr Nd145

FIG. 2. Decay chains involving krypton and xenon.

krypton and xenon which diffuse out of the fuel elements during criticality dissolve in the water. However, after the water is dumped following the shut-down, the krypton and xenon have the ability to diffuse through the fuel elements to the atmosphere and their daughters cause the aerosol contamination detected. A compilation of the decay chains involving krypton and xenon is shown in Fig. 2.[1] Since the daughters of Kr^{85} and Xe^{133} are stable, these noble gases do not contribute to the measured aerosol activity. The daughter of krypton-87 has a long half-life (10^{10} years) and cannot contribute significantly to the radioactivity of the measured aerosol. The same is true of the daughters of Xe^{135} and Xe^{137}. Any noble gas with a half-life shorter than 5 sec probably cannot diffuse from the fuel elements to an appreciable extent before decaying to a metal atom. This eliminates daughters of krypton-92, 93, 94, 95, 97 and xenon-141, 143, 144 and 145. The remaining isotopes to be considered are krypton-88, 89, 90, 91 and xenon-138, 139 and 140. It is the decay products of these isotopes which will be important toxicologically and which will be collected by air sampling media.

Fig. 3 shows the rate of build-up of selected isotopes of krypton and xenon and the relative concentrations of these isotopes after 2 hr of critical operation.[2] It should be noted that the half-lives of krypton-90, krypton-91, xenon-139 and xenon-140 are so short that they decay rapidly following shut-down. Krypton-89 has a high equilibrium value and persists following shut-down. Although krypton-88 does not approach its equilibrium value in a 2 hr operating interval, its long half-life of 2.8 hr results in its continuous release from the fuel elements for several hours after shut-down. Xenon-138 reaches its high equilibrium value and will escape from the core for an appreciable period of time following shut-down, resulting in a high concentration of 32 min Cs^{138} in the air.

Chemical analysis of collected air samples has borne out the supposition that noble gases are the precursors of the airborne activity. From 95 to 100 per cent of the activity collected on a Millipore filter has followed rubidium and cesium in chemical extractions. The measured half-life of the rubidium activity was about 18 min, corresponding to rubidium-88, rubidium-89 and rubidium-91. The half-life of rubidium-90 is too short to allow detection by the methods employed. The measured half-life of the cesium activity was 30 min, corresponding to cesium-138 and masking that of cesium-139.

On one occasion, following a particularly high flux level in a plastic fuel critical assembly, a 51 day half-life activity was found in trace quantities. This was probably strontium-89 which has a half-life of 51 days, and which has been found in small amounts in the water surrounding the core.

Detectable amounts of Sr^{90} have not been found. It should be noted that its precursor, krypton-90, has

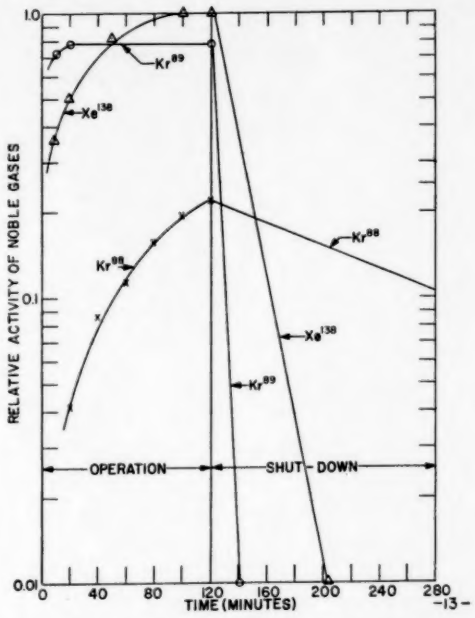


FIG. 3. Relative activity of selected noble gas fission products as a function of reactor operation time.

such a short half-life (33 sec) that its equilibrium value is not high in the operating core. That which is produced decays rapidly following shut-down so that only a small amount of krypton-90 escapes from the fuel elements. Further, the ratio of the half-lives of krypton-90 (33 sec) and strontium-90 (28 years)[3] indicates that the amount of krypton-90 which does escape from the core will produce insignificant strontium-90 activity in the air.

It was concluded that the aerosol consisted mainly of a mixture of Rb^{88} and Cs^{138} with small quantities of Rb^{89}, Cs^{139}, and Ba^{139}.

Figure 4 is a plot of aerosol concentration (on a log scale) as the ordinate versus the time after shutdown of the critical assembly. The symbol T_0 was the time the water was emptied from the pot, exposing the fuel to the air. The upper curve is typical for operations using the plastic fuel. This particular curve shows that the concentration of airborne activity at the top of the pot was approximately 3000 d/min per m^3 (1.5×10^{-9} μc/ml) 20 min prior to shutdown, i.e.

while the assembly was critical and the fuel was covered with water. The airborne activity increased quite rapidly within 2 min after dumping the water and reached a maximum of 7 to 8 $\times$ 10^6 d/min per m^3 (3.5 $\times$ 10^{-6} μc/ml) about 1½ hr after shutdown.

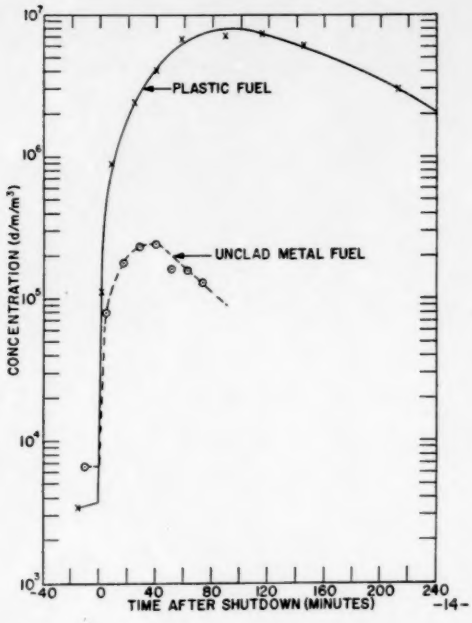


FIG. 4. Variation in aerosol activity from operation of experimental critical assembly.

It then decreased to insignificant levels within a few hours. Concentrations as high as 43 $\times$ 10^6 d/min per m^3 (2 $\times$ 10^{-5} μc/ml) have been measured over plastic fuel reactors. The lower curve is typical of operations using an unclad metal fuel. Here the airborne activity increased from approximately 6000 d/min per m^3 (3 $\times$ 10^{-9} μc/ml) prior to shutdown to a maximum of 250,000 d/min per m^3 (10^{-7} μc/ml) about ¾ hr after shutdown. Again, the increase was quite rapid as soon as the water was dumped exposing the fuel to the air. However, the maximum concentration of airborne activity arising from the use of unclad metallic fuel was much lower than for the plastic fuel; in this case they differed by a factor of 30. The time that the maximum concentration occurred after shutdown has varied considerably and was influenced by factors associated with the escape of the contaminated air from the open top of the pot (room air currents, etc.).

The toxicology of the aerosol found in the assembly room has been evaluated in terms of the exposure to the lung. The lung exposure criteria was selected on the strength that the major components present are of short half-life and will dissipate most of their energy in the lung. It is possible to use a number of different assumptions concerning the behavior of the aerosol to arrive at a maximum permissible concentration (MPC). We have chosen to use the MPC resulting from calculation of the continuous occupational exposure to each isotope that will yield 300 mreps/week to the lung, and then select the most stringent condition. On this basis the MPC has been determined as approximately 20 $\times$ 10^6 d/min per m^3 (9 $\times$ 10^{-6} μc/ml).

In practice, it has been found that the inhalation dose is small compared to the external exposure present in the critical assembly room following operation. The MPC of the aerosol to yield 300 mreps/wk whole body occupational exposure for immersion in an infinite cloud is approximately 2 $\times$ 10^6 d/min per m^3, only one-tenth of the inhalation MPC. To be added to this external immersion exposure is the direct radiation from the core. Therefore, it is seen that the inhalation toxicity of the aerosol is a minor part of the total exposure to personnel.

Since the personnel receive some external radiation exposure, both from the noble gases and other external sources, it has been our practice to reduce the personnel exposures to the aerosol either by use of filter-type respirators or exhaust ventilation. A slot-type hood was built on the periphery of the assembly tank and, when operating with an exhaust rate equivalent to 50 ft^3/min per ft^2 of tank surface, the concentration of the aerosol in the air above the rim of the tank has never exceeded 10^6 d/min per m^3 (5 $\times$ 10^{-7} μc/ml) or about 5 per cent of the MPC. This exhaust system is more effective when started at approximately the same time that the water is dumped. Its operation at the time when the highest concentrations normally occur has been effective in controlling the potential hazard. The exhaust air from this hood is filtered through a high efficiency dry-type filter before discharge outside the building.

Tests on the MSA Ultra-Filter Respirator have been made by passing contaminated air through the respirator filter in series with a Millipore filter, with the air flow being maintained approximately at breathing rates. Efficiencies of the respirator filter material have been about 93–95 per cent. Overall efficiency of the half-mask filter respirator would probably be about 90 per cent where care is taken in its fitting and use. This degree of protection is considered adequate for this application, and respirators are used when exhaust ventilation is not provided.

It was concluded that the radioactive aerosol from an experimental critical assembly consisted principally of Rb^{88} and Cs^{138}. The mechanism by which the aerosol was produced was explained by the ability of the noble gas precursors, primarily Kr^{88} and Xe^{138}, to escape from the fuel elements. Local exhaust ventilation or the use of filter-type respirators provided adequate personnel protection.

REFERENCES

1. C. D. Coryell and N. Sugarman (Editors), *Radiochemical Studies: The Fission Products*, NNES. McGraw-Hill, New York (1951).
2. J. R. Donnelson, *Theoretical Study of Fission Products Gaseous Activity From a Homogeneous Reactor*. LRL-153 (1954).
3. W. H. Sullivan, *Trilinear Chart of Nuclides*. Oak Ridge National Laboratory (1957).

Health Physics Pergamon Press 1958. Vol. 1, pp. 141–151

ASSESSING THE IMPACT OF ENVIRONMENTAL RADIATION ON TOTAL HEALTH*

PAUL S. HENSHAW
Division of Biology and Medicine, United States Atomic Energy Commission, Washington, D.C.

(*Received* 15 *November* 1957)

Abstract—Ionizing radiation is but one of various agents that affect the stamina and well-being of population groups within species through induction of mutations in both the somatoplasm and the germplasm. A preliminary effort has been made to identify and examine central and pace-setting determinants of the life process, moving thereby toward a better understanding of the role and significance of radiation.

On the basis of accumulated information and deductions that follow logically, questions have been raised regarding (1) the part natural radiation may have had in shaping the character of life processes through time, possibly influencing both the course and rate of evolution and the average life-span of species, (2) the matter of whether radiation, though injurious to cells, may not under certain circumstances have a beneficial effect in complex multicellular organisms, (3) radiation as an agent acting to reduce the fitness reserve in organisms, and (4) the limits of exposure so far as individual and group survival are concerned.

A formulation has been developed which aids in visualizing the "total health" process and how this is affected positively and negatively by various factors, including radiation. By means of the formulation some impression is gained of the stability of the life process, on the one hand, and, on the other, of its extreme vulnerability if an environmental factor such as radiation is altered significantly.

PART 1

BEING aware that environmental radiation of the ionizing type is increasing on a global scale, and that such radiation affects the behavior of various life processes and biologic systems, it is a matter of importance to go as far as possible in evaluating such effects on total health and well-being. It is also a matter of importance to make some appraisal of health conditions expected when different levels of environmental radiation prevail—that is, when exposures are in the range of present background or natural levels, when they become increased moderately by industrial and other applications, and when they might be increased very greatly by nuclear warfare.

For the purposes here, we shall employ the World Health Organization definition of health —*complete physical, mental and social well-being and not the mere absence of disease.* Among the many factors in addition to infectious diseases that affect total health are: nutrition, toxic products, native abilities and skills, social customs, reproductive practices, selection processes, availability of resources, etc., including environmental radiation. As in any such complex process, there are central and pace-setting components that act to determine the rate of change or the conditions of equilibria, and to the extent that such determinants can be identified and measured, there is opportunity to relate and weight representative elements in a formulation and thereby to obtain an index of the overall situation. With the aid of such a formulation, it should be possible, depending on assumptions, definitions and measurements, to obtain meaningful impressions concerning the extent and relative significance of radiation as a factor influencing general well-being.

With the present state of knowledge, it will not be possible to go nearly as far as desired,

* Paper presented at the Second Annual Meeting of the Health Physics Society, Pittsburgh, Pa., 17–19 June 1957

either in distinguishing the central determinants of total health, nor in relating and weighting such elements accurately. However, enough information is available to enable some delineation of the problem and to make progress in developing ideas about limits and danger points relative to the life mechanism—particularly with respect to radiation exposure. After presentation of certain background information, the concepts and some details of different approaches will be outlined and applied to the extent possible.

The life process

Both individuals and populations (thinking primarily of higher animal forms) have as the basis for their existence certain biochemical processes (mainly oxidation-reduction) which operate within or in close association with cells—mechanisms that utilize nutritional and other elements as a source of energy to keep in motion the reactive and adaptive activity identified as living. In general, it can be said that so long as there is a suitable environment and a continuous supply of nutritional fuels, the energizing process known as life can go on. Of importance here are the facts that the energy mechanisms (i.e. the oxidation-reduction process) are much alike, if not actually identical, in different species, and that it is the materials of inheritance—the autosynthesizing nucleic acids (DNA, RNA) associated especially with chromosomes—that act to determine the particular pattern of use to which the energy mechanism is put; that is, whether it is to operate in a manner characteristic of one species or another, of one strain or another, or indeed whether it will operate efficiently. Said in another way, chromosomal materials, by means of the energy directing process, are regarded as responsible not only for species and strain differences, but also to a large degree for differences in stamina and vitality within species and strains.

Biologic activity

The pattern of energy utilization and the consequent perpetuation of life, both in individuals and in population groups, can be modified by affecting the supply of raw materials (that is, the nutritional elements), by affecting the functional parts of cells (meaning the cytoplasmic and various nonhereditary parts —the machinery, so to speak, by means of which raw materials are utilized), and by affecting the hereditary materials themselves (the guiding or governing components). Radiation as an agent produces ionization and excitation in each of these elements. Such changes culminate in loss or alteration of critical materials and in increase of decomposition products. These effects, in turn, lead to slowing of mitosis, to accumulation of toxic products and to cell death (usually at the time of the next cell division), any of which can be expected to interfere with the well-being of organisms exposed to radiation. Loss or alteration of critical materials can mean reduction in concentration of enzymes, hormones or other biocompounds, and they can mean alteration of the autosynthetic nucleoproteins of chromosomes—the hereditary components—which influence so strongly the specific function of cells. All of these effects are known to follow irradiation and important biological changes have been shown to stem from each. A substantial body of evidence, however, tends to indicate that the mutational type changes (chromosomal and genic) are especially significant inasmuch as most other effects are transitory, and inasmuch as lethal and sublethal mutations have a strong influence on the existing generation and sublethal have such an influence on subsequent generations. It is necessary, therefore, to recognize mutational changes as one important means if not the main means by which radiobiologic modifications are induced.

In the same way that individuals are dependent upon the flow of chromosomal materials from one cell generation to the next in growing tissues, so also are populations dependent on the flow of such materials from one generation to the next; it is in the composite or aggregate of chromosomal substances of the soma and the germline cells of populations that one finds the substances which comprise the basis for individual and population group existence and the seat of general weakness or strength, as the case may be. In dealing with the effects of environmental radiation on total health, we are concerned with effects on the germplasm as a

single and independent entity, though spread throughout the population.

Genetic quality

From experience with breeds of livestock and strains of laboratory animals, it is known that populations can be vigorous or weak, healthy or ill, in much the same manner that individuals can be strong or sickly. Since the raw materials used in the life sustaining process are essentially the same, and since the energy process itself is much the same even in widely different species, it is necessary to conclude that breeds and strains, as well as stamina and vigor, are dependent upon genetic quality.

Parenthetically, it must be said that for purposes of this paper, the term "genetic" is applied rather more broadly than sometimes is done. When used here, it will pertain to chromosomal as well as genic features, and to perpetuation of hereditary materials in the somatoplasm as well as in the germplasm. Similarly, when the expression "gene pool" is used, the meaning intended will be aggregate of chromosomal materials and not aggregate of genes alone. The broader definitions are regarded as necessary because of the significance of chromosome aberrations as well as gene mutations, and because of the importance of hereditary materials in functions of the soma as well as functions of the germplasm.

Genetic quality, as is well known, can be changed by the action of mutagens—including radiation which is now a well recognized mutagen. Other mutagens are heat and chemical compounds of many types. Significantly, the chromosomal aggregate, or the gene pool, for somatic cells in individuals and that for germline cells in populations, appear to be affected in the same way by radiation, but the biologic consequences differ due to variation in functional operation of the two cell types.

Mutagenic action

Considering the mechanics and also the consequences of mutagenesis, it should be remembered that all three types of mutation—*lethal*, *deleterious* and *beneficial*—are induced in both somatic and germline cells. Lethal mutations, either of the somatic or germline variety, terminate in cell death, thus, in themselves, exerting no further influence on the organism. Cell loss, however, may be severe enough to impair organ function or cause organ failure, and organism death as a consequence. Complete destruction of stem cells or sterilization of organs whose function in considerable measure is to proliferate and differentiate cells required for special functions may also occur.

Deleterious mutations, being scattered at random in the growing tissues and existent in cells capable of further mitosis, have somewhat different effects, depending on whether they are in somatic of germline cells. In the soma, it is obvious that the mutated cells which proliferate must give rise to less efficient cellular components, leading thus to gradual tissue transformation and to reduced organ efficiency. On the basis of proliferative behavior, it is apparent that such changes culminate eventually in organism death when transformation has progressed to a point that failure of a vital organ occurs. In the germplasm, deleterious mutant cells multiply in essentially the same manner, but important points of difference in later behavior are that through breeding, mutant cells become spread in populations, and selection processes operate, discriminating in favor of cells better able to perform functions required by host organisms. With respect to selection, there is a weeding-out or discrimination that occurs in connection with mating and also in connection with fertilization. Of importance here is the fact that selection of the same type can not take place in the case of somatic mutations.

Beneficial mutations, though comparatively very infrequent in both the germline and the soma, appear to perform a particularly significant role in germline materials, but no such role in the somatic. In case of the germplasm, beneficial mutations are retained by selection, thus being accumulative through succeeding generations and contributory to the progress of evolutionary development. In the soma, on the other hand, beneficial mutations obviously cannot pass from one generation to the next (in sexual forms), and since the number produced in any one individual or in any one generation

is inconsequential, they exert no significant effect.

Regenerative action and equilibria

Thus far, it has been stressed that lethal mutations obviously have their greatest influence in terms of cell losses in both somatic and germline cells leading thereby to impairment and failure of organ function in the soma and to failure of reproduction as a feature in connection with the germplasm. An important feature in the general picture is that the regenerative function—the ability of tissues to replenish themselves—acts to counteract cell losses such as those caused by continuously acting mutagens, including environtal radiation. It is evident that conditions of cellularity at any particular time are determined by the rate at which cell loss is offset by regenerative activity.

In case of the soma, it is apparent that if destruction of cells is at a low rate, replenishment will be sufficient to keep pace with loss, the result being that no cell deficit or organ failure is seen; on the other hand, when destruction is at a high rate, regenerative capacity of the tissue or organ becomes exceeded with the result that cell deficit and organ failure in some situations are evident. In this connection it is important to remember that, during the normal course of life, cells are used up or consumed continuously as body functions are carried out and that the losses are counteracted by generative growth with the result that cell content of tissues and organs tends to remain constant within certain general limits. When a few more cells are consumed, as in case of low level irradiation, proliferation simply proceeds at a slightly accelerated rate in order to compensate for the losses. Clearly, however, compensation can be complete only as long as rate of destruction does not go beyond regenerative capacity.

Some idea of regenerative power can be gained by recalling that laboratory animals can be bled sufficiently to cause generation of a full body content of blood every 2 to 3 weeks without causing noticeable change in the peripheral picture, or by recalling that four-fifths of a rat liver can be extirpated and full replacement seen in a few weeks' time. What has just been said with respect to cell loss stemming from lethal mutations is in contrast with the situation that develops as a consequence of sublethal mutations where there is continuous accumulation of less efficient deleterious mutant cells rather than replacement of destroyed ones. With respect to the somatoplasm, therefore, there exists cell loss and replacement going along with lethal mutations, and accumulation of cells able to survive but less useful to the host organism going along with the sublethal. Limits of the first come in terms of regenerative capacity, and limits of the second in terms of generalized reduction of tissue efficiency, culminating eventually in failure of one or another of the vital organs.

Interestingly, a somewhat analogous situation exists in connection with the germplasm, but the principles involved have different meaning and the consequences are different. By considering particular features of the two patterns, it is possible to recognize central characteristics of the life process and central elements of the radiation health hazard problem at the same time.

With deleterious mutations occurring more or less continuously in germline cells, and with continuous selection in favor of the cells best suited for carrying out functions necessary for perpetuation of the species, there is compensatory elimination of the least useful mutants. In a sense, the compensatory elimination here is comparable to regenerative replacement of lost cells in the somatoplasm in that critical levels are maintained. With opposing forces operating in the germplasm, an equilibrium becomes established as in case of the somatoplasm. When the rate of mutagenesis is low, the equilibrium becomes stabilized with the load of deleterious mutations at lower levels, and when high it becomes stabilized with the load at higher levels. Since mutagenesis goes on continuously (irrespective of man-made radiation) it is evident that a quantity of deleterious mutations is carried continuously by the germplasm. From the obvious evolutionary progress, it is apparent that some burden of deleterious mutations can be carried and a reasonably high standard of health and well-being maintained at the same time. Indications are that as the load of

deleterious mutations is increased, stamina and vigor of the population group decreases—such decrease being made evident by increased morbidity and by shortened life tenure—the late effects of irradiation exposure, including those that may occur in subsequent generations. The effect is one of reducing resiliency or narrowing the margin of safety with respect to generalized ability of populations to resist disease and overcome adverse conditions of all types.

In light of background information thus developed, the problem becomes one of evaluating the inter-related radiobiologic effects recognized and of orienting these in the total health picture. Steps will be taken in this direction by means of four approaches: (1) giving some figures for radiation mutagenesis, (2) giving some figures for irradiation injury, (3) extrapolating from the mutagenic effects of natural radiations, and (4) (Part II) utilizing a special formulation to obtain an index of total health for population groups.

Radiation mutagenesis

Estimates based on extensive analytical studies suggest (they do not fully establish) that 30 to 80 rep per generation applied to populations will double the frequency of mutations.* Doubling the frequency of beneficial mutations conceivably could double the rate of evolutionary progress, but since this moves so slowly under normal circumstances, the change would probably be imperceptible during a single lifetime of observation. Similarly, because of the remarkable regenerative capacity associated with most growing tissues, doubling the loss due to lethal mutations occurring normally, would probably be undetectable. Doubling the burden of deleterious mutations, however, is a different matter. An index of damage caused by deleterious mutations (according to a special report of the National Academy of Science)[1] "is the totality of tangible genetic defects in living individuals" (mental defects, epilepsy, congenital malformations, neuromuscular damage, hematological and endocrine abnormalities, modification in vision and hearing, cutaneous and skeletal abnormalities, and injury to the gastro-intestinal and genito-urinary tracts). It has been estimated that 4 to 5 per cent of all live births in the United States have defects of this sort, and of these about half (2 per cent of the total live births) have simple genetic origin and appear prior to sexual maturity.

According to the special report, if the human species were continuously subjected to a doubling dose of radiation, the present level of 2 per cent of such genetic defects might be expected to rise, eventually reaching the doubled level. With respect to the next 100,000,000 births in the United States (approximately the number of children to be born to the presently alive population) something like 2,000,000 (2 per cent), according to the report, would experience genetic defects of the sort listed—these resulting from the deleterious "spontaneous" mutant genes induced by means other than man-made radiations. Should the population be subjected generation after generation to a mutation doubling dose of man-made radiation, also according to the report, the present figure of 2,000,000 would gradually rise to 4,000,000, perhaps 10 per cent of which, or some 200,000, would occur during the first generation. These figures refer to ideas about tangible inherited defects, not including inherited tendencies for diseases of different types. In considering a twofold increase in deleterious mutations, we are confronted with the question of whether addition of such a burden is tolerable. Some orientation may be provided by remembering that traffic fatalities in the United States are running between 35,000 to 40,000 annually—the price paid for having automotive transportation.

Irradiation injury

Further insight into the irradiation injury problem is gained by considering the following somewhat rounded figures:

1.0 rep/day†—the threshold level of dose for detectable injury in individual human beings in 2 to 3 years.

0.1 rep/day—the accepted permissible dose during the 1930's—at that time a level believed to be non-injurious.

* See The National Academy of Science Summary Report on "The Biological Effects of Atomic Radiation" 1956 for review!

† Exposure doses measured in air.

0.01 rep/day—recently advocated permissible level (with some qualifications).
0.001 rep/day—natural radiation.

With respect to acute exposures, the following figures are of significance:

50 to 100 rep—probably not noticed in any way by subjects, though detectable microscopically in peripheral blood.
100 to 200 rep—some radiation sickness, but no deaths.
200 to 400 rep—severe radiation sickness; death in less than half the subjects exposed.
400 to 800 rep—severe radiation sickness; death in more than half the subjects exposed; death in essentially all when exposure is as much as 800 rep.

One rep per day, according to these figures, is in the threshold range of dose deleterious to the well-being of individuals of the human species. One-tenth rep per day, which at an earlier time was regarded as "safe", is now held in question due to greater awareness of possible reduced vigor caused by an increasing burden of deleterious mutations. One one-hundredth rep per day is at least a more conservative figure irrespective of other things to be said about it; this figure is especially interesting, however, because it is only a factor of ten above natural radiations, and therefore raises the question about the attitude which should be taken toward the background radiations that have been continuously a part of the environment, probably longer than the presence of life on the earth.

Earth and cosmic radiations

To the extent that natural radiations ionize and excite atoms and molecules, it must be said that they are like other radiations in affecting living systems. Limited, and as yet very preliminary, considered estimates indicate that from 10 to 20 per cent of the mutations which occur "naturally", are attributable to radiations coming from the earth and cosmic sources. As yet no reasons have been developed for expecting that natural radiations act differently from other ionizing radiations in causing mutations—that is, with respect to number induced per unit dose, types induced, and proportions of the types induced.

A point basic to present considerations is that life arose and has evolved to its present status in different species in the face of naturally occurring radiations, and that the resiliency and stamina of organisms have been sufficiently great to compensate for existent mutational handicaps and to support evolutionary progress such as it has been. On the one hand, it is possible to visualize how mutagenesis has contributed constructively to evolutionary progress, and, on the other, how it may have tended to reduce species vigor and resiliency, thereby contributing to a lessened efficiency for the maintenance of life. It thus becomes necessary to ask regarding the level of environmental radiation which is tolerable and needed as well as that which is intolerable, in the general interest. In asking such questions, we are leading toward a philosophy and general attitude about irradiation exposure. In reality, we are asking what level of exposure is best, all things considered—what level provides the greatest good for the greatest number in the long run. Granting that natural radiations may have had a constructive role in the life process, we are asking whether any increase above this level must necessarily be regarded as deleterious.

PART II*

Index of total health

"Radiomimetic" is a term applied to agents that act similarly to ionizing radiations in affecting living processes. Many of the appurtenances of civilization, and especially those associated with technological advancements of the past few decades, are known to affect genetic quality. Various drugs, including medicinals, insecticides, plant stimulants, preservatives, dyes and other products, have been shown to have mutagenic effects. Such agents, along with ionizing radiations, are contributing to the burden of deleterious mutations in the germplasm. Other agents—specifically, practices, in this case—contribute to the burden by reducing the rate at

* The ideas on bioeconomic progress or total health as given here represent the combined thinking of Dr. ALEXANDER LANGER and the author. Development of the concept was done largely before the author came to occupy his present position.

which mutations are eliminated from the germplasm by the selection processes. The practice of modern medicine and modern public health, sacrifice of the more fit in warfare, and differential acceptance of fertility control among social classes, are means by which the burden of deleterious mutations is kept greater.

With the multiplicity of existent radiomimetics, with the possibility that the action of any one of these—particularly radiation—might be increased very significantly (as in the case of nuclear warfare), and with the fact that still other types of agents and factors act strongly to influence the condition of total health and well-being, there is need to understand the effects of radiation against the total backdrop of agents and factors influencing general health. A start can be made toward achieving these objectives, by utilizing an approach developed in preliminary manner for considering standards and levels of living in different countries and territories.[2,3,4,5]

"Standards and levels of living", as used earlier, was intended to have essentially the same meaning as the expressions "total health" or "total well-being" employed here. The approach used formerly was to assume that total well-being is the consequence of a supply and demand relationship in which members of population groups compete for the available goods needed for living. According to this idea, the competition is influenced by population pressure on the one hand, and by per capita productivity on the other—a ratio of the two providing a situation index by means of which actual and potential well-being of different population groups can be compared.

Of the many elements regarded as representative of per capita productivity, three were chosen for use: caloric intake per person per day (C), industrial energy per person per year (E), and income per person per year (I). Units were selected with particular points in mind and the elements were multiplied to provide necessary acceleration. The expression CEI then gave an arbitrary value which was used as a numerator. Population pressure was considered to consist of inhabitants per unit land area (N) multiplied by an expression for full fertility potential defined in a special way ($2^P + 2^D$) involving the interaction of birth and death rates, P being natural population increase rate and D being crude annual death rate. The expressions (2^P) and (2^D) are geometric ones, such being needed here to cover the compounding features of growth involved. When P is 1, the population doubles in about 70 years and when it is 2 it doubles in about 35 years, etc., which is characteristic of the human species. The expression (2^D), as used here, represents lost fertility, and accordingly must be added to (2^P) to give full fertility potential. The expression $N(2^P + 2^D)$ was used as a denominator, and the complete formulation was:

$$\frac{CEI}{N(2^P + 2^D)} = ITH \text{ (index of total health, or situation index).}$$

The formulation, the choice of terms, and the weighting of terms, as used, represents a first attempt to develop a situation index for comparing conditions of health and well-being in different countries and territories, or in population groups. Improvements of the formulation are to be expected with time, but dealing with it in its present form helps considerably in understanding that radiation is but one of the many factors influencing the life process—that is, influencing the underlying energy mechanism consisting mainly of oxidation-reduction to follow the course it does follow.

By using figures set forth in the previous papers for different countries and territories (representative of the period around 1950), and plotting the individual values for caloric intake, industrial energy consumed, annual income, birth rate and death rate, a graph is obtained as shown in Fig. 1. The curves in the upper part of the graph can be said to be representative of economic and technical progress and those in the lower part to characterize vital behavior. The whole graph can be referred to as a picture of bioeconomic progress or of progress in total health. Basic data, it should be emphasized, were stable enough to establish general trends only and not detailed ones.

As countries and population groups advance economically and technically they progress from left to right on the graph, certain shifts in vital behavior helping to determine rate of movement

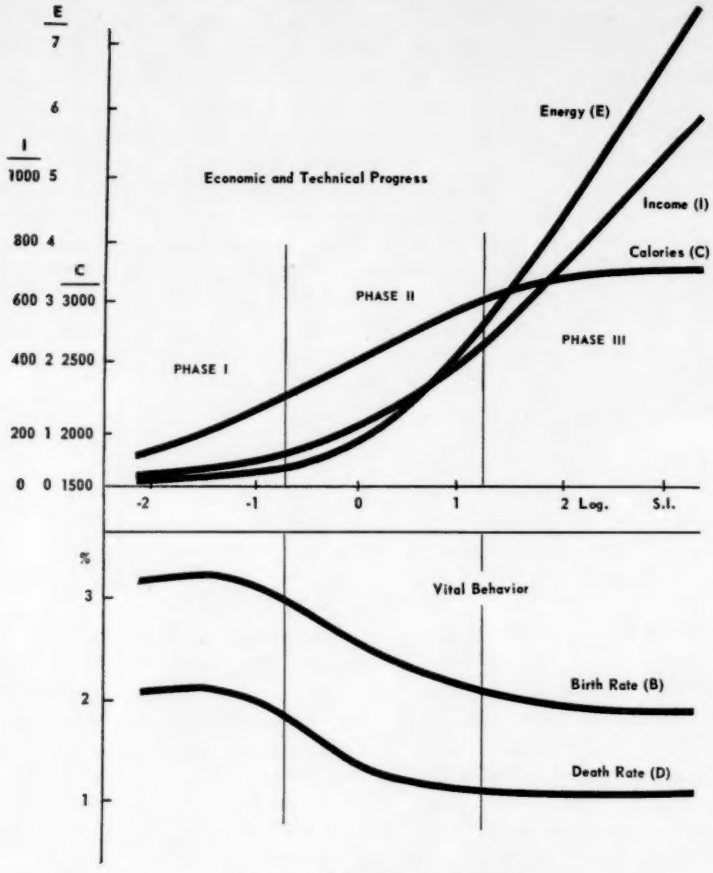


FIG. 1. Bioeconomic progress.
I—Per capita income per year in U.S. dollars equivalent.
E—Industrial energy consumed per person per year in tons coal equivalent.
C—Caloric intake per person per day.
S.I.—Situation index.

and the direction. For convenience, the development can be divided into three phases: Phase I being characterized by low nutrition, high birth and death rates, and low industrial and earning activity, Phase II by transition in all of these aspects, and Phase III by high nutrition, low birth and death rates, and by high and accelerating industrial and earning activity. Of particular importance here is the fact that countries and population groups can move to the left as well as to the right on the graph.

With the formulation giving support to ideas that maintenance of population groups is a dynamic process involving central and pacesetting determinants, and with the graph giving a picture of the process, further inquiry can be made about the significance of radiation in the general picture.

At the outset, it is evident that according to this approach there are no factors in the formulation for radiation as such. This means, if the formulation is regarded as a satisfactory representation of the entire bioeconomic process or of total health, either that radiation as a general agent affecting life is thought of as secondary to

or represented by other elements, or as inconsequential. This situation is explained by remembering that until recently, radiation has been essentially a constant element in the environment like time, space and gravity, therefore requiring no special attention. Now, however, with environmental radiation changing measurably and with the possibility that this element could change markedly under certain circumstances, particular consideration is called for and it is important to ask additionally what the role of radiation has been and whether it could become a central determinant.

According to the picture of bioeconomic development as set forth, it is important—if not actually necessary—to look upon the life process as existing fundamentally in terms of concentrations of people and of their abilities and skills for developing and utilizing the things needed for life—elements of the denominator and numerator, respectively, in the formulation. The main question then is how radiation acts in relation to these components. Asking the question in this way makes evident, perhaps more forcefully than by other means, that radiation is but one of a great number of factors that can and do act to affect the basic life process or the basic force for life—also that radiation under certain circumstances could easily become a dominant factor, if not the one main controlling factor. From the information assembled, it is evident that radiation affects both the numerator and the denominator in the situation index formula. Through increase in the availability of nuclear energy for industrial purposes and through increase in the availability of specific radiations for medical and other purposes, the opportunities for earning a livelihood are enhanced; also, through increase in the amount of stray radiation the hazards for life are enhanced, thereby affecting death rates. The two types of action are antagonistic and of course it is the net result that determines gain or loss so far as opportunities for adaptation and survival are concerned.

When radiations are employed in such a way as to increase productivity—that is, by increasing the output of things needed for life—conditions of living and the total health improve, and when they result in a reduction of stamina and vigor of individuals and thereby of population groups, conditions of living and the total health deteriorate.

In some respects it would be desirable at this point to insert various test values for irradiation injury into the formulation in order to develop impressions about magnitudes of exposure required to bring down the level of living by significant increments—or to bring it up. Many elements act together in determining the abilities and skills and thereby the productivity of population groups; these include availability of resources, condition of health, education and various others. Similarly, many elements act together in determining population density, including nutrition, social customs, climate and disease. Radiation is but one of the many factors functioning to affect the different central determinants.[6] Separating the elements and placing a value on each cannot be done with a significant degree of accuracy at the present time, yet very often correlations can be made between measurable modifications of particular features in the total health mechanism and particular aspects of the environmental milieu. Consideration of the elements and subelements in a total picture aids greatly in avoiding prejudicial and distorted weighting of components.

SUMMARY AND COMMENTS

Based on the concept of total health, and on mutagenesis as a key determinant, the following thoughts and impressions emerge:

(1) Mutagenesis, which appears to have been a fairly uniform factor in nature throughout the history of living things, is being accelerated in recent decades by human practices stemming to a large degree from scientific and technical advancements.

(2) Ionizing radiations are but one of the many elements of modern living which induce mutations, and mutagenesis is only one of various processes that affect the life mechanism of species.

(3) Recent investigations of radiation mutagenesis have been instrumental in drawing attention to changes in genetic quality as an important element in modern living. With concentration of attention on reduction of species

vitality caused by radiation, it is important to maintain perspective by keeping in mind other agents that have similar and possibly greater effects.

(4) Indications are that natural radiations coming from the earth and cosmic sources, and man-made radiations coming from particle accelerators and from radioactive materials, affect living things in the same ways—among them mutagenesis involving the lethal, deleterious and beneficial type mutations.

(5) Cell losses such as may result from lethal mutations are compensated by regenerative function in organisms, and lowering of vitality of the germplasm such as may result from deleterious mutations is compensated by selection processes; similar lowering of vitality in the somatoplasm is not compensated by the same kind of selection process.

(6) In the total life mechanism of species, the lack of opportunity for selection against deleterious mutations in the soma can be regarded as a means by which the struggle for life in individuals is overcome leading thereby to death as an inevitable consequence, whereas the opportunity for such selection to take place in the germplasm can be regarded as enabling continued species life.

(7) Accepting that mutagenesis contributes to aging of the soma, and that the natural life tenure of human beings (barring fatal disease and accidental death) is about 70 years, it appears that natural radiation may be a component involved in fixing this figure.

(8) Accepting also that the present status of species development constitutes evidence of evolutionary advancement, it follows that the burden of deleterious mutations in the germplasm, including those induced by natural radiations, has not been sufficient to prevent significant evolutionary progress. *A priori* evidence makes clear that the regenerative and selective functions that are operative in nature have been adequate in surviving species, including human, to compensate for critical cell losses that may have occurred, and for the general devitalization stemming from mutagenesis—including the 10 to 20 per cent believed to result from natural radiations.

(9) Species vitality is measurable most readily in terms of morbidity, longevity and reproductive ability. In the case of human beings, longevity is probably the most reliable measure inasmuch as people, almost universally, try to live as long as possible. Morbidity is also a useful measure insofar as statistics are adequate and reliable. Reproductive ability probably has little meaning however since few parents struggle to have the maximum number of children.

(10) From figures available (mainly for animals), it is evident that life shortening effects caused by irradiation of the soma alone are detectable when doses are in the range of 1 rep/day. When effects on the germplasm—that is, those coming from ancestry—become added to those induced in the soma during any immediate generation, it is to be expected that life shortening or an increase in morbidity would be detected for lower levels of exposure of the soma; as yet, however, research results are not sufficient to give answers on this matter.

(11) From the bracketing provided by knowledge of the threshold level of dose for detectable hematological and histological damage resulting from changes in the soma alone, and of the level of background radiation which is accepted as tolerable—it appears likely that the threshold of injury for combined somatic and germplasm effects would fall in the range of 0.01 to 0.1 rep per day.

(12) It is an axiom of biology that adaptation of a species to its environment is accelerated by increase in diversity of types of individuals in a population such as occurs with increase of mutations, and on such a basis it can be postulated that the best biological adaptation to a radiation environment would come from an increase in mutation rate, such as caused by radiation itself. But, it is also an axiom of biology that a more rapid adaptation achieved by this means must be accompanied by an increase in percentage of defectives in the living population. The matter thus comes to the question of relative values—the point of the greatest good for the greatest number in the long run.

(13) With a protective environment, involving antibiotics, medical care, and elaborate public health measures, we are, among other complex features, confronted with the question of whether

the genetic armament required must be as strong, and whether it may not be acceptable in the general interest to carry a somewhat heavier load of mutations in the population as a means of gaining adaptation advantages.

(14) The various points developed here, give confidence in the point of view that the 2 to 3 per cent increase in background radiation provided by fallout from nuclear tests would be so small that it would be imperceptible in the welfare of the human race—but, it gives a basis for very grave concern if exposure becomes some thousands of times greater than background such as would be expected in the case of nuclear warfare.

REFERENCES

1. *The Biological Effects of Atomic Radiation.* Summary Reports from a Study by the National Academy of Sciences National Research Council, Washington, 1956.
2. A. LANGER and P. S. HENSHAW, *Amer. J. Med. Sci.* **230,** 119–127 (1955).
3. A. LANGER and P. S. HENSHAW, *Amer. J. Med. Sci.* **230,** 484–490 (1955).
4. A. LANGER and P. S. HENSHAW, *Amer. J. Med. Sci.* **230,** 605–621 (1955).
5. A. LANGER and P. S. HENSHAW, *Amer. J. Med. Sci.* **231,** 407–426 (1956).
6. H. B. JONES, *Estimation of effect of Radiation Upon Human Health and Life Span.* Proceedings of the Health Physics Society. (1956). Available from R. G. GALLAGHER, *Tres., Health Phys. Soc.*, 42 Shirley Rd., Wellesley, Mass.

Health Physics Pergamon Press 1958. Vol. 1, pp. 152–163

SOIL DISPOSAL OF RADIOACTIVE WASTES AT ORNL: CRITERIA AND TECHNIQUES OF SITE SELECTION AND MONITORING*

K. E. COWSER and F. L. PARKER

Oak Ridge National Laboratory, Operated by Union Carbide Nuclear Company, for U.S. Atomic Energy Commission, Oak Ridge, Tennessee

(*Received* 16 *December* 1957)

Abstract—The ability of soil to remove and retain the bulk of the fission product cations from aqueous solution is used as the basis for disposal of large volumes of intermediate level wastes. An experimental system consisting of three 1 million gal disposal pits has been developed at ORNL and through December 1956, received a total of 5.6 million gal of waste containing 58,000 c of Cs^{137}, 15,000 c of Ru^{106} and lesser amount of Sr^{89}, Sr^{90}, Co^{60}, Sb^{125} and the rare earths. Sodium ion and nitrate ion account for the largest part of the chemical constituents.

The geologic and hydrologic characteristics of the reservation served as a guide in the selection of small areas for detailed study. The information obtained from field studies of a specific area and laboratory investigations of the interaction of waste and typical soil were used to estimate the operating characteristics of a disposal pit.

After the pit was in operation, the underground movement of the chemical constituents and radionuclides in the waste was followed by sampling and radiologging observation wells and sampling at stream gauging points in the pit area. These results have confirmed the preoperating estimate of the direction and path of waste movement and the seepage of anion materials such as Ru^{106} and NO_3^- through the soil.

INTRODUCTION

THE use of soils for the disposal of radioactive wastes has been practiced at the Oak Ridge National Laboratory since the beginning of operations. Contaminated solids are disposed of by burial in unlined trenches that are capped with an earthen or concrete cover. At the present time the burial operation is utilizing land at the rate of 5 acres/year.[1] Since 1952, intermediate level-radioactive liquid wastes have been discharged to the soil by means of surface pits. Although the purpose of this paper is to present some of the more significant findings in the disposal of liquid waste, the discussions of criteria and techniques of site selection and monitoring have application to the disposal of solid wastes as well.

The types of liquid wastes produced by a nuclear power industry are related to their source. These wastes are the result of feed material processing, reactor operation, reactor fuel processing, equipment decontamination, and laboratory operations. High level wastes are generally associated with reactor fuel processing, and an estimate of their volume and fission product concentration can be made by considering the probable growth of the industry. As an example, it has been estimated that 1.5×10^{12} kWh/year of electrical energy will be generated in the year A.D. 2000.[2] Consequently, about 250 metric tons per year of fission products will be produced in a volume of 5.6×10^7 gal/year.†[3] Even after 100 days of cooling, each gallon of high level waste will contain from 80 to 5200 c of hazardous fission products.[4] Low level wastes and intermediate

* Paper presented at the annual meeting of the Health Physics Society, 17–19 June 1957, Pittsburgh, Pennsylvania.

† This volume is based upon an irradiation level of 4000 MW-days/ton of uranium and the processing of 190 metric tons/day at a volume of 800 gal/metric ton.

level wastes may be identified with all of the sources listed. The distinction between low and intermediate level waste streams is that low level waste can be released to the environment without exceeding the maximum permissible concentration for fission products in drinking water and/or air (10^{-7} and 10^{-9} μc/ml of mixed fission products, respectively). Although the biologically hazardous strontium-90 and cesium-137 will be small contributors to the total activity in the intermediate level waste, the level of beta, gamma, or alpha activity present will not permit direct discharge to the environment. The volume of low and intermediate level waste produced can be expected to exceed that of high level waste.

Soils have the ability to remove fission products from aqueous solution by ion exchange, by adsorption, and by filtration. Ionic species of the fission products, and especially the cations such as cesium and strontium, may be exchanged for the natural constituents of the clay particle fraction of the soil. The capacity of a soil for the removal of specific radionuclides is influenced by the type of ion and its valence state, the concentration of the ion, and the salt concentration and chemical composition of the waste solution. Radionuclides in colloidal form may be adsorbed and particulate material filtered from solution as the waste passes through the soil. The attractiveness of soil disposal for low and intermediate level waste disposal is further enhanced by the large areas and volumes available in most locations, the varied methods available for their use, and the lower cost when compared with tank storage. Present day soil disposal systems bring the liquid waste in contact with the soil by seepage from open pits and cribs. The sudden release of large quantities of radioactive waste into the environment from a properly located and managed soil disposal system is virtually impossible in comparison to the failure of a storage tank by rupture.

ORNL Experience with Soil Disposal of Liquid Wastes

Site selection

Proper selection of a site for waste disposal requires advance study of the environmental factors that may affect the construction of the plant or be affected by the operation of the disposal system. This is because the environment will influence the type and cost of the facility, will establish operating limits for the system, and will partially determine the type and location of monitoring facilities.

Environmental considerations have been enumerated by THEIS,[5] GORMAN[6] and MORTON.[7] To determine if a given site is suitable for disposal of waste into the ground, the geology, hydrology, meteorology, and use of natural resources in the area must be investigated. More specifically, the cost of construction, stability, and extension of the system, will be influenced by the depth and character of the overburden, the elevation at which rock may be encountered, and the structural integrity and areal distribution of the formation. The acceptance of liquid by a formation and the rate, direction and extent of underground movement of waste constituents is related to the physical and chemical properties of the formation such as the porosity, directional permeability and ion exchange capacity, as well as localized recharge from rainfall and the slope of the ground water table. Since ground water can convey fission product contamination to surface streams by basal flow, the points of ground water discharge must be established and the dilution capacity of the surface streams determined. The ability of the atmosphere to dilute gaseous or particulate contamination resulting from waste disposal operations is determined by wind speed and direction, frequency and duration of temperature inversions, and the distribution of rainfall.

Only after investigation of the above factors can a realistic evaluation of the suitability of the site for disposal of various levels of fission product wastes be completed.

ORNL environment. The Laboratory lies in the midst of the southern section of the Ridge and Valley Province (a subdivision of the Appalachian Highlands Division), which is characterized by alternating and parallel ridges and valleys that trend northeast in accordance with the strike of the underlying rocks. In all, the reservation covers an area of 90 miles^2. Oak Ridge is the nearest community and lies 15

miles from the Laboratory. The nearest surface water supply is 6 miles downstream from the Laboratory on the Clinch River and furnishes water to another AEC installation. Ground water users in the area are essentially nonexistent, with the nearest known and operating private supply located 6 miles from the Laboratory.

The general geology and ground water hydrology was investigated by geologic mapping, core drilling, and measurement of ground water levels.* The Oak Ridge area is underlain by four sedimentary formations (Rome formation, Conasauga shale, Knox dolomite, Chicamauga limestone) which have been folded and faulted. Valleys have been developed in the belts of less resistant rock and the ridges are underlain by more resistant strata. In general the beds dip about 30° to the southeast, but minor folds are common, particularly in the incompetent shales of the Rome and Conasauga formations.

The Laboratory (Bethel Valley) is underlain by the Chickamauga limestone, which contains minor beds of shale, siltstone, and clayey limestones with associated chert. Ground water movement is primarily along bedding planes and fractures. Some of the fractures are slightly enlarged by solution; nevertheless the porosity is low. Immediately to the north (Chestnut Ridge) thick beds of calcium–magnesium carbonate of the Knox dolomite are predominant. Ground water in the Knox moves through solution cavities, some of which are of cavernous proportions. To the south, Haw Ridge is underlain by the steeply dipping, Rome sandstone, which is fine grained, shaly, and well cemented. Water movement is largely confined to openings between the beds. The permeability and porosity are small. The Conasauga shale in Melton Valley overlies the Rome formation, which at its base is a sandy-silty shale, and grades through calcareous shale, interbedded limestone and shale, and terminates at the upper end in masssive limestone beds.[8]

The ion exchange capacity of the calcareous shale, in which the waste pits are excavated, is about 28 m equiv/100 g. BROCKETT[9] has shown that this shale can remove the important radioisotopes of cesium, strontium, and the rare

* The current exploratory and descriptive geologic and hydrologic investigation at ORNL is carried out by the U.S. Geological Survey.

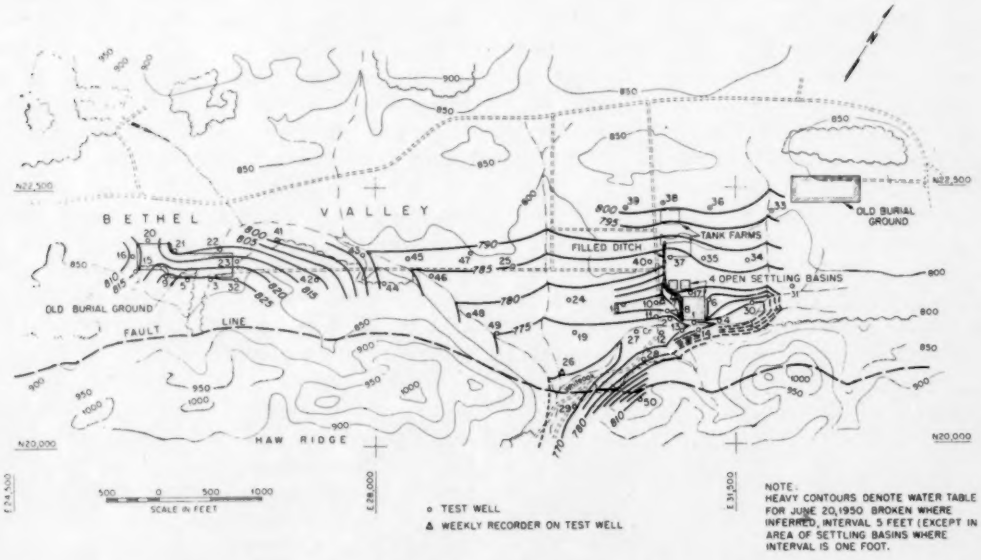


FIG. 1. Surface topography and water table elevations of the Oak Ridge National Laboratory area.

earths from aqueous solution. More recent column studies with ORNL wastes have shown that the shale can remove about 135 μc/g of cesium before any cesium can be detected in the effluent.[10] For disposal purposes at ORNL, the Conasauga shale of Melton Valley is the most promising formation on the basis of its geologic, hydrologic, and chemical characteristics as well as its geographic location.

Water level measurements in the Laboratory and Melton Valley areas have revealed that the ground water level generally conforms with surface topography, which indicates that it occurs under water table rather than artesian conditions. This is indicated in Fig. 1, where the water table contours have been superimposed on a topographic map of the section. These contours are suppressed replicas of the topography. During wet seasons of the year the water table rises and intercepts the ground surface at low points forming springs and seeps. Since the slope of the water table influences the rate and direction of ground water or waste movement, water level measurements are made to determine the hydraulic gradients.

The ability of a formation to accept liquid waste and the rate and direction of underground movement of these wastes are further investigated by detailed drilling in the area of interest, pressure tests, pumping tests, and tracer tests. By these techniques the Conasauga shale in Melton Valley was determined to be somewhat weathered to a depth of from 10 to 30 ft depending on location. Although the ability of this shale to transmit water is low by water supply standards,* the permeability appears to be more uniform than that of the other formations. By introducing water under pressure into open wells (pressure testing), a few thin zones have been located below the weathered shale that accept water at the rate of about 1 gal/min. At depths below 80–100 ft the shale is impermeable essentially. Consequently, most of the liquid waste seeping from a shallow pit would flow through the weathered shale.

By withdrawing known quantities of water from a center well in a group of wells and observing the drawdown in the other wells, the hydrologic parameters which govern partially the rate and direction of ground water flow can be investigated. These pumping tests have revealed that the direction of the strike has a pronounced influence on the rate and direction of water movement. This direction, therefore, would be critical in the selection of pit sites.

Because pumping tests have limited usefulness in non-homogeneous aquifers, testing a pit with water before introducing actual waste will provide valuable information on the hydrodynamics of an operating pit. A small experimental pit 20 ft diameter by 5 ft depth constructed in the Conasauga shale was so tested. In Fig. 2(a) water table contours are shown as they existed before the test began, while Fig. 2(b) shows the effect of recharge. It can be seen that waste movement from the pit will be in the direction of wells U4 and U16 primarily, and, to a lesser extent, toward well U2. By tracing the stable nitrate in a 1.6 molar acid aluminum nitrate waste solution, the path of waste movement was confirmed as shown in Fig. 2(c). More precise predictions of pit operation would be possible if the rate of seepage were determined and an appropriate tracer were used to determine the average velocity of water movement through the ground. These data combined with water level measurements can be used to estimate the porosity and permeability of the formation.

The local topography exerts some control on the prevailing wind direction, which is usually either up-valley from the west to the southwest or down-valley from the east to the northeast. However, the ridges have had no significant effect on precipitation or cloudiness. Atmospheric diffusion in the ORNL area has also been studied.[11] For the 12 years 1945–1956, an average precipitation of 52.55 in/year has been recorded.[12] Evaporation stations in East Tennessee, employing standard U.S. Weather Bureau evaporation pans, have recorded a mean annual evaporation of 41 to 43 in/year.[13]

* Most water-bearing materials utilized by wells have coefficients of permeability between 10 and 5000 according to MEINZER. O. E. MEINZER, *Hydrology* (1st Ed.) pp. 452–455. Dover Publications, Inc., New York (1949).

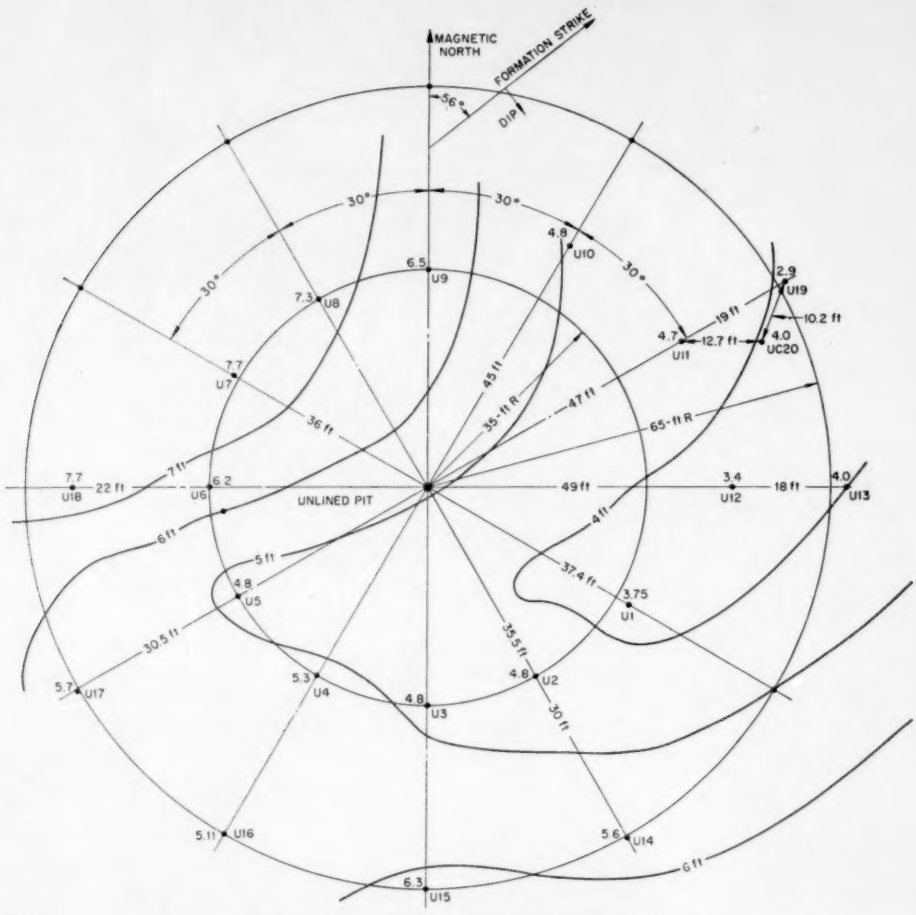


FIG. 2(a). Experimental pit: normal water table contours.
● Observation and sample wells (results of 19 May 1954).
—— Relative water-table contours to common base.

Monitoring

An estimate of the operating characteristics of a waste disposal system can be made on the basis of presite testing and laboratory experiments. The accuracy of the estimate depends in large measure on the complexity of the environment. Consequently, control points are required that will provide information on the volume and composition of the waste and the dispersion of the waste after it is released to the environment.

Description of ORNL waste pits. A system of surface pits has been constructed in the Conasauga shale for the disposal of intermediate level fission product wastes. This method of disposal is based on the concept that as the waste seeps slowly through a large volume of shale, the shale will retain enough of the radioactive constituents to prevent the widespread dispersion of hazardous amounts of activity in the environment. The pits are located in the White Oak Lake drainage basin in which surface runoff and basal flow are collected and monitored before the liquid leaves the controlled area.

There are three 1,000,000 gal pits in operation,

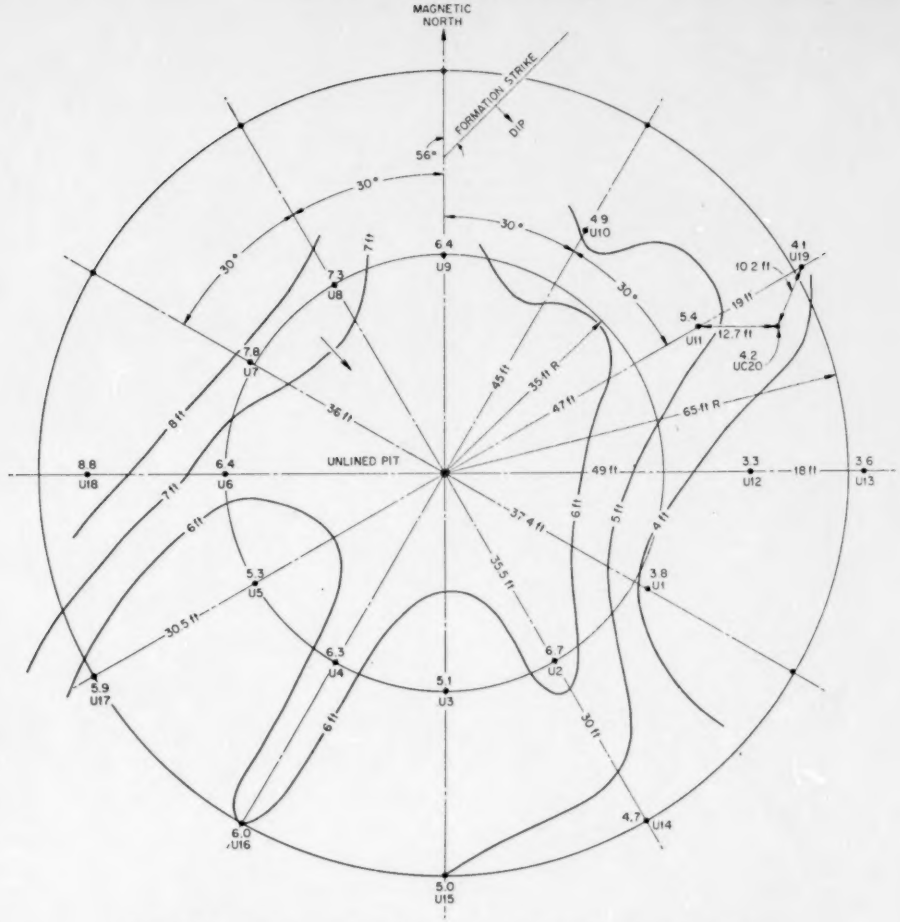


FIG. 2(b). Experimental pit: water table contours after recharge.

● Observation and sampling wells (results of 2 June 1954).

—— Relative water-table contours to common base.

each similar in design. An end view of Pit No. 3 is shown in Fig. 3. The pit has the shape of a frustrum of an inverted rectangular pyramid, 210 ft × 100 ft in top plan and 15 ft deep. A 1 in. mesh galvanized iron screen covers the pit to prevent animals from gaining access to the waste. Liquid levels are determined periodically by means of a staff gage, and continuous records of liquid stage can be maintained through use of the available stilling well and "float" recorder.

The layout of the pits is shown in Fig. 4. The present operating procedure is to pump the waste from the Laboratory through 7000 ft of 2 in. welded steel pipeline direct to Pit No. 3. Valved overflow pipes connect the pits in series and allow waste to be overflowed into Pit No. 2 and Pit No. 4. The numbered circles show the location of wells that were installed to provide geologic and hydrologic information, and to detect the underground movement of waste.

Volume and composition of waste. The volume of waste transferred to the pits is determined by measuring the amount of liquid pumped from the collection tank.* The volume of waste

* An underground storage facility at ORNL provides eight concrete gunite tanks with a total capacity of 1,105,000 gal.

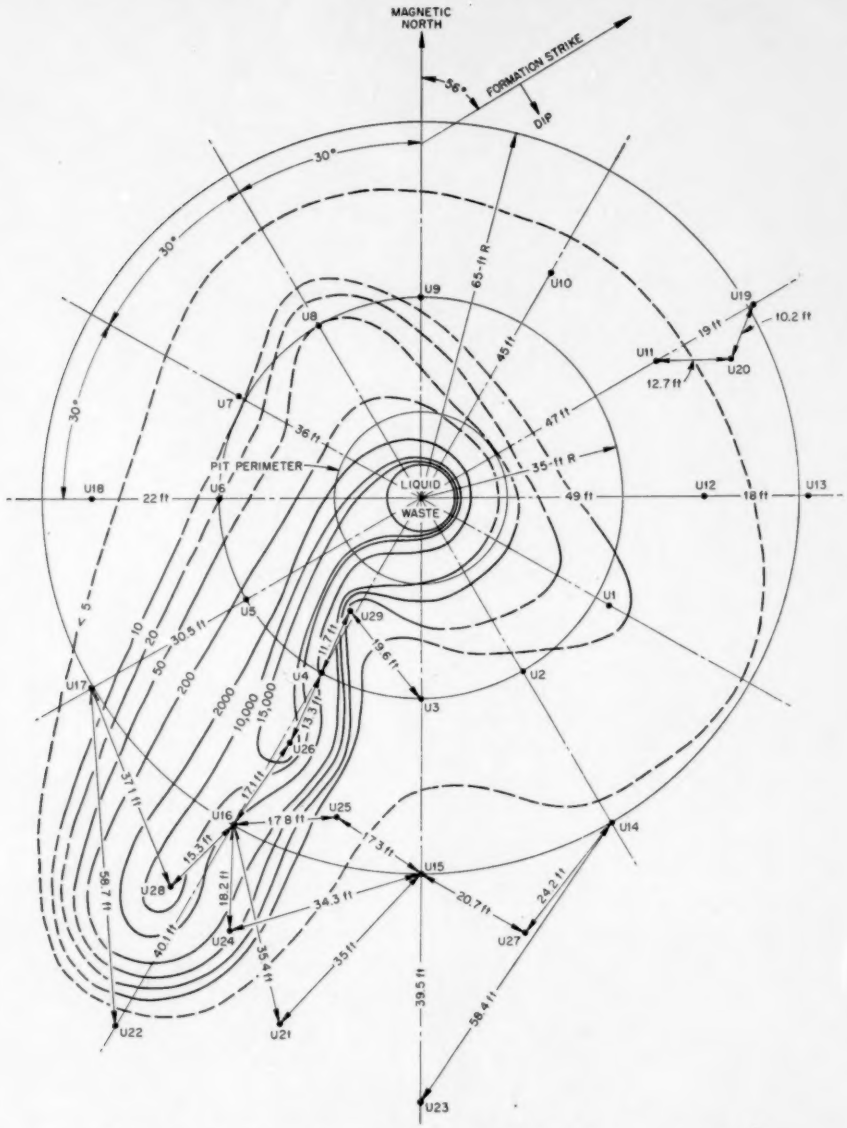


FIG. 2(c). Experimental pit: dispersion of non-radioactive wastes.

● Observations and sampling wells (results of 13–14 September 1954).

—— Nitrate contamination contours in p.p.m. based on field and laboratory analysis—location approximate.

—— Nitrate contamination contours—location inferred or projected.

overflowed between pits is based on staff gage readings before and after the overflow period and pit rating curves that equate liquid stage and volume. Composite samples collected during the transfer or overflow period are analyzed for specific chemical ions and radionuclides.

The accumulated volumes and major fission products released to the pits from June 1952 through December 1956 are summarized in

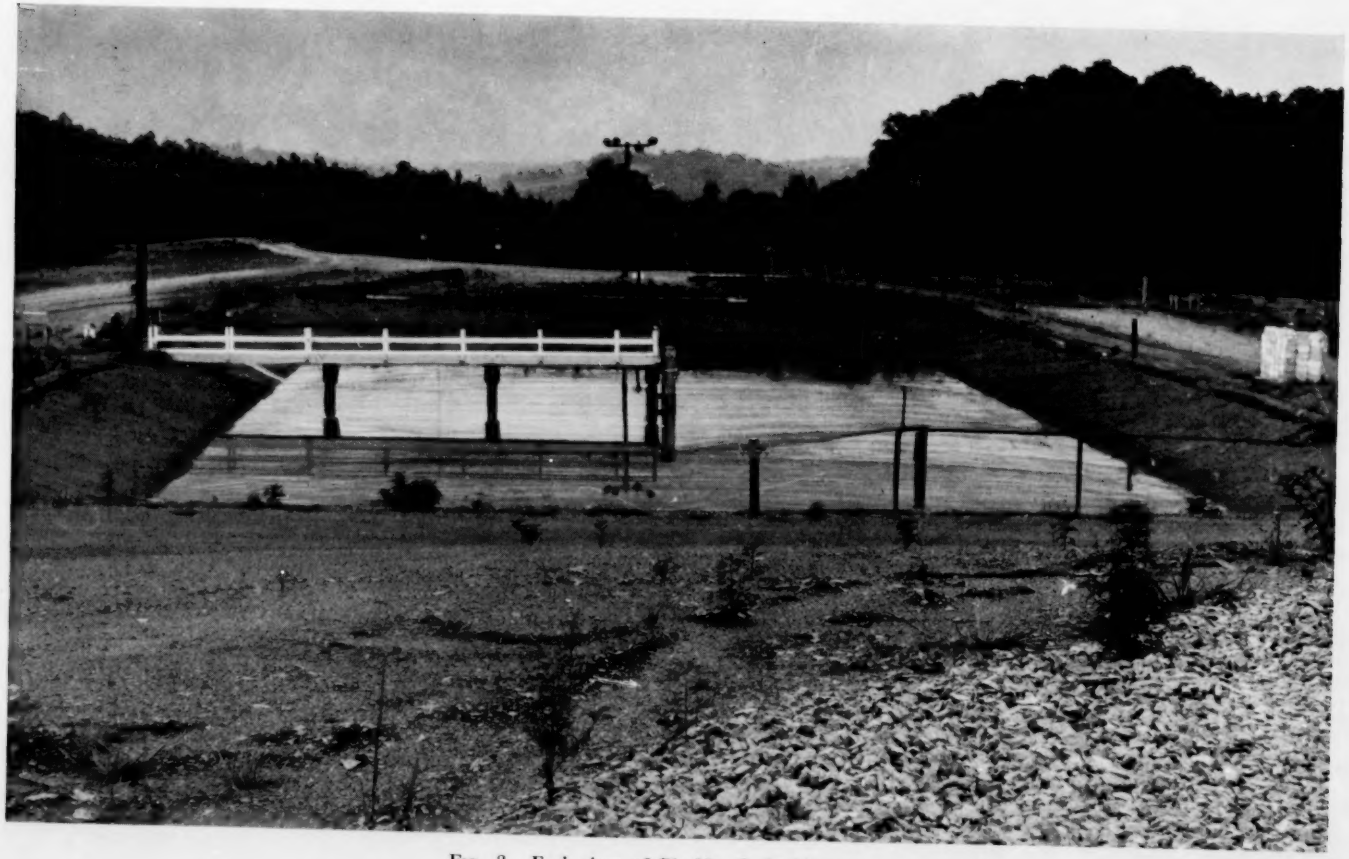


FIG. 3. End view of Pit No. 3, looking west.

FIG. 5. Well-logging units.

Table 1. The inventory of individual pits separates the events of direct waste transfer and overflow. From the total inventory, 5,600,000 gal of waste containing 58,000 c of cesium-137, 15,000 c of ruthenium-106, and lesser amounts of rare earths, strontium,

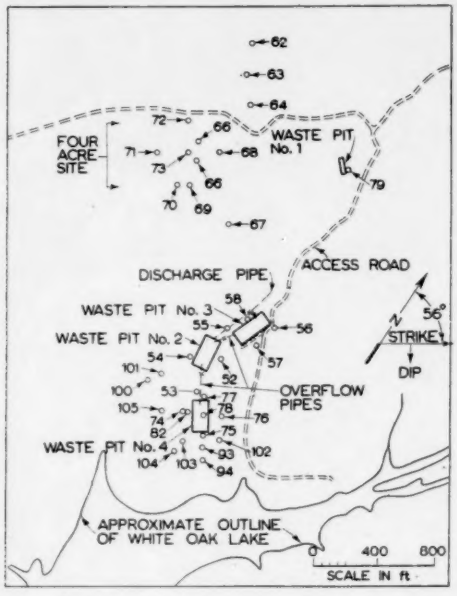


FIG. 4. Waste pit disposal area.

cobalt and antimony have been released to these disposal pits. On the average, cesium and ruthenium account for 90 per cent of the activity in the waste, with the exception of periods when special programs or operating procedures alter the composition. During such periods iodine-131 and the rare earths are the major contributors. The normal waste stream contains an average of 16 mg/ml of sodium and 24 mg/ml of nitrate, and they account for 70 per cent of the total solids. In excess of 160 tons of sodium and 220 tons of nitrate have been released.

During the course of waste transfer from Pit No. 3 to Pit No. 4, the concentration of fission products in the waste is reduced. The amount of cesium and ruthenium reduction is 75 to 80 per cent. The decrease in fission products is the result of their occlusion in the precipitates settling in the first two pits, adsorption on the side walls of the pits, dilution by rainfall, and radioactive decay.

Liquid loss from the pits. Loss of liquid from the pits is an important consideration in the control of the system and forecasting future requirements. An adequate number of pits must be available to balance the loss of liquid with the volume of input. The design of pits is also influenced by this factor.

Since it is not possible to measure directly the rate of seepage from the pits, seepage must be determined as the net loss after accounting for waste pumpage, rainfall and evaporation. It has been shown that the volume of rainfall entering a pit can be estimated within 3 per cent by simply considering this volume to be that which is added by rain falling within the catchment area of the pit. Analysis of evaporation pan and meteorological data from the waste pit and Laboratory areas have been completed by the U.S. Weather Bureau.[14] The use of nomograms devised by KOHLER[15] were instrumental in this analysis. It has been shown that comparable results are obtained by estimating pit evaporation from evaporation pan and meteorological data from the pit area, while the use of Laboratory meteorological data provides estimates only slightly less reliable.

The rate of seepage from a pit is influenced by the liquid stage in the pit, the depth to the ground water table around the pit, the viscosity of the liquid waste, and the permeability of the soil. The estimated rate of seepage from Pits No. 2 and No. 3 is shown in Table 2. It is observed that while the rate of seepage from Pit No. 2 increased each year, the rate of seepage from Pit No. 3 decreased in 1956 despite a consistently higher pit stage. This is believed due to the raising of the water table around Pit No. 3, and particularly to the northeast, resulting in lower water table gradients. The seepage rate of both pits appears to increase during the warm weather and decrease with the colder weather. Although the rates of seepage for separate years cannot be compared directly, due to increasing pit stage and changes in water table conditions, there is nothing that suggests a marked change in the permeability of the soil. The liquid level in Pit No. 4 falls at nearly a constant rate of

*Table 1. Radionuclides transferred to waste pits individual pit inventory**

	Pit No. 2				Pit No. 3				Pit No. 4			
		Curies				Curies				Curies		
	Gal ($\times 10^3$)	Beta ($\times 10^2$)	Cs^{137} ($\times 10^2$)	Ru^{106} ($\times 10^2$)	Gal ($\times 10^3$)	Beta ($\times 10^2$)	Cs^{137} ($\times 10^2$)	Ru^{106} ($\times 10^2$)	Gal ($\times 10^3$)	Beta ($\times 10^2$)	Cs^{137} ($\times 10^2$)	Ru^{106} ($\times 10^2$)
Waste pumped to pit	1241	160	142	65	4400	548	435	84	—	—	—	—
Waste overflowed to pit	3163	89	109	21	—	—	—	—	1308	23	27	6
Waste overflowed from pit	−1308	−23	−27	−6	−3163	−89	−109	−21	—	—	—	—
Net waste to pit	3096	226	224	80	1237	459	326	63	1308	23	27	6

*Total system inventory**

	Gal ($\times 10^3$)	Curies		
		Beta ($\times 10^2$)	Cs^{137} ($\times 10^2$)	Ru^{106} ($\times 10^2$)
Pit No. 2	3096	226	224	80
Pit No. 3	1237	459	326	63
Pit No. 4	1308	23	27	6
Total	5641	708	577	194

* All amounts are accumulative through December 1956.

0.21 ft/day irrespective of liquid stage, and is due almost entirely to seepage.

It is of interest that although the liquid loss from Pit No. 3 during 1956 just balanced the influx due to rainfall, the 310,000 gal of waste that seeped out of the pit carried with it appreciable amounts of radioactive material. Thus the volume of waste overflowed to Pit No. 2 remained the same but the concentration of fission products and chemicals was reduced; consequently the burden on the absorptive capacity of the soil around this pit was lessened.

Dispersion of waste. After a pit has received radioactive liquid waste the underground movement of stable chemical ions and radioactive ions is followed by well sampling and well-logging in the pit area. The wells are sampled at pre-selected depths, based in part on logging results. Fig. 5 shows the well-logging units now in use. The trailer mounted probe consists of a halogen type Geiger-Muller (GM) tube* and a preamplifier mounted in an aluminum housing. The scintillation probe on the right, designed at NYO-AEC,† includes a $1\frac{1}{8}$ in. diameter by $1\frac{1}{2}$ in. thick sodium iodide crystal, phototube, preamplifier, and high voltage supply in a stainless steel housing. The other scintillation probe, designed by the

* Anton Electronic Laboratories, Inc., Brooklyn, N.Y. Effective cathode dimensions: Small tube $\frac{9}{32}$ in. diameter by $1\frac{9}{16}$ in. long; Large tube: 0.60 in. diameter by $5\frac{15}{16}$ in. long.

† U.S. Atomic Energy Commission, N.Y. Operations Office, 70 Columbus Ave., New York, N.Y.

USGS,* contains a similar crystal and phototube in a brass housing, with the remaining electronics located in the count rate meter. Calibration of the scintillation and GM probes

Table 2. Estimated seepage from Waste Pits 2 and 3

Date	Pit 2		Pit 3	
	Seepage (gal/day)	Pit Stage (ft)	Seepage (gal/day)	Pit Stage (ft)
1953	1900	3.7	—	—
1954	2700	8.3	—	—
1955	3400	9.9	1650	9.2
1956	3850	12.2	950	12.7

in air gives the curves shown in Fig. 6. With present circuits the USGS probe (high voltage of 1130 V) is the most sensitive and the 3 in. GM tube operates most satisfactorily at higher radiation intensities. However, calibration in air is not indicative of the sensitivity in well water. By calibrating the equipment in the field in a 6 in. diameter aluminum container (sunk in the soil) filled with solutions of ruthenium-106–rhodium-106 in equilibrium, it was determined that the sensitivity of the two scintillation probes and the large GM tube was 1×10^{-5} μc/ml and the small GM tube about 3×10^{-4} μc/ml.

Lateral movement of wastes. Samples collected from test wells provide the necessary data to construct a map of the concentration of radioactivity in the pit area, as shown in Fig. 7. Within the accuracy of present analysis techniques, ruthenium-106 is the primary radionuclide involved in underground travel with lesser amounts of the complexed ions cobalt-60 and antimony-125 present. Nitrates are predominant in the contaminated wells. Activity has spread principally in an east–west direction or along the strike of the formation and confirms the findings of presite testing. The rate of underground waste movement in this direction from Pit No. 4 is from 10 to 30 ft/day, and from Pits No. 2 and No. 3 about one-fifth as great. Little movement has occurred across the strike, south of Pit No. 4. These facts reflect the inhomogeneity that exists in this geological formation, even within short distances. The activity concentration lines terminate at two small streams, east and west of the pits and represent the development of surface seeps at low points in topography.

Weirs were installed last year at the streams in the pit area and continuous records of stream flow were obtained. From these records it has been observed that the concentration of activity in the streams fluctuates rapidly and depends on weather conditions and the flow in the stream, while the total transport per unit time does not change markedly but does increase during the early period of surface runoff when the stream beds are washed out. During the last 4 years an average of 23 c/year of ruthenium-106 entered the White Oak Creek–Clinch River system, presumably due to operation of the waste pits. Assuming complete dilution in the Clinch, the average concentration of ruthenium-106 is about 10^{-8} μc/ml.

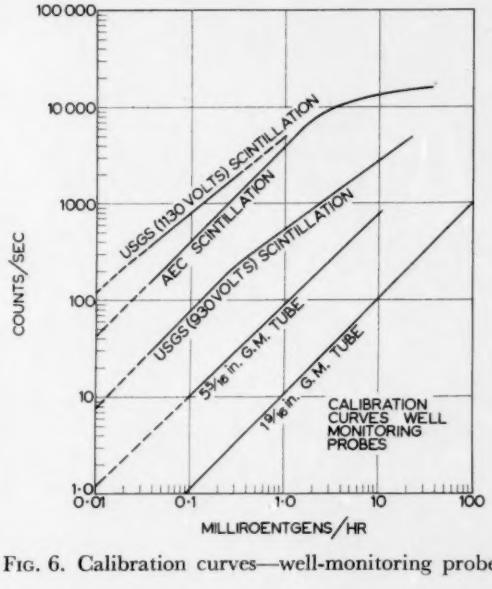


FIG. 6. Calibration curves—well-monitoring probes.

* U.S. Dept. of Interior, Geological Survey, Federal Center, Denver, Colorado.

Vertical movement of waste. The vertical pattern of waste movement from the pits is traced by radiologging test wells. From these logs the original entrance of radionuclides to the well can be defined, but as the concentration of activity in the well increases, the interpretation of the logs becomes complex. The counting rate is influenced by (1) activity in the well water, (2) variations in the diameter of the test holes, (3) adsorption of activity on the side wall of the hole, (4) sorption of activity in the rock, and (5) activity in solution in the strata through which wastes are traveling. To isolate these variables, the contaminated liquid of several wells was mixed to provide a uniform concentration of activity and was subsequently replaced with tap water to eliminate this source of radioactivity. Following each of these procedures the well was logged. The results showed that zones of greatest activity were associated with the sorption of activity on the side wall of the hole and/or in the rock formation.

Two grouted wells were installed to assist in this definition of underground movement. One well is located in the center of Pit No. 4 (Well No. 82) and the other well 75 ft west of the pit (Well No. 82); both wells were completed before any waste had been released to the pit. A grouted well, shown in Fig. 8, is an open boring in which a well casing is centered and sealed with cement grout. Radionuclides moving through the formation in the vicinity of these wells can be detected and located with the gamma sensitive well probes, without entering the well and complicating the radiologs.

Typical logs of the grouted wells and an open boring 95 ft west of the pit (Well No. 74) are shown in Fig. 9. A continuous zone of activity had penetrated to a depth of 6 ft below the bottom of the pit after 4 weeks of operation, but had ceased upon reaching the more consolidated shale at 20 ft. Most of the radioactive cations had been removed within the top 12 in. of shale. The major portion of the activity intercepted by the grouted well adjacent to the pit occurs in discrete zones down to a depth of 40 ft. The trace of the counting rate in the open boring is similar to that of the grouted well to this depth; below 40 ft, it is difficult to determine whether or not the waste is entering the well in zones marked by increased activity.

Airborne contamination and external radiation. Although cesium-137 has not reached the test wells in detectable quantities, it has been

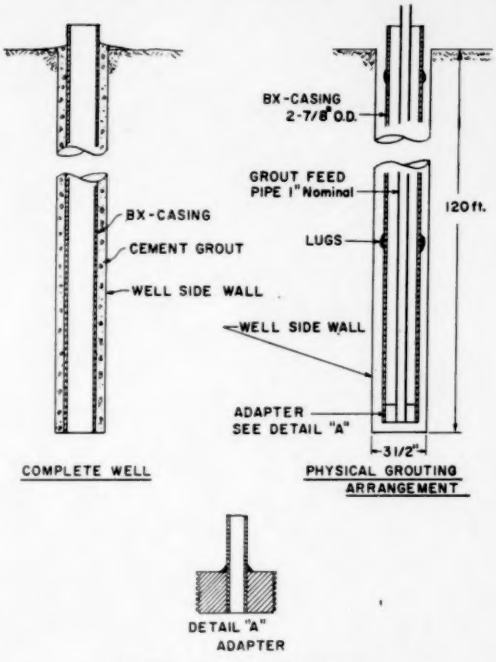
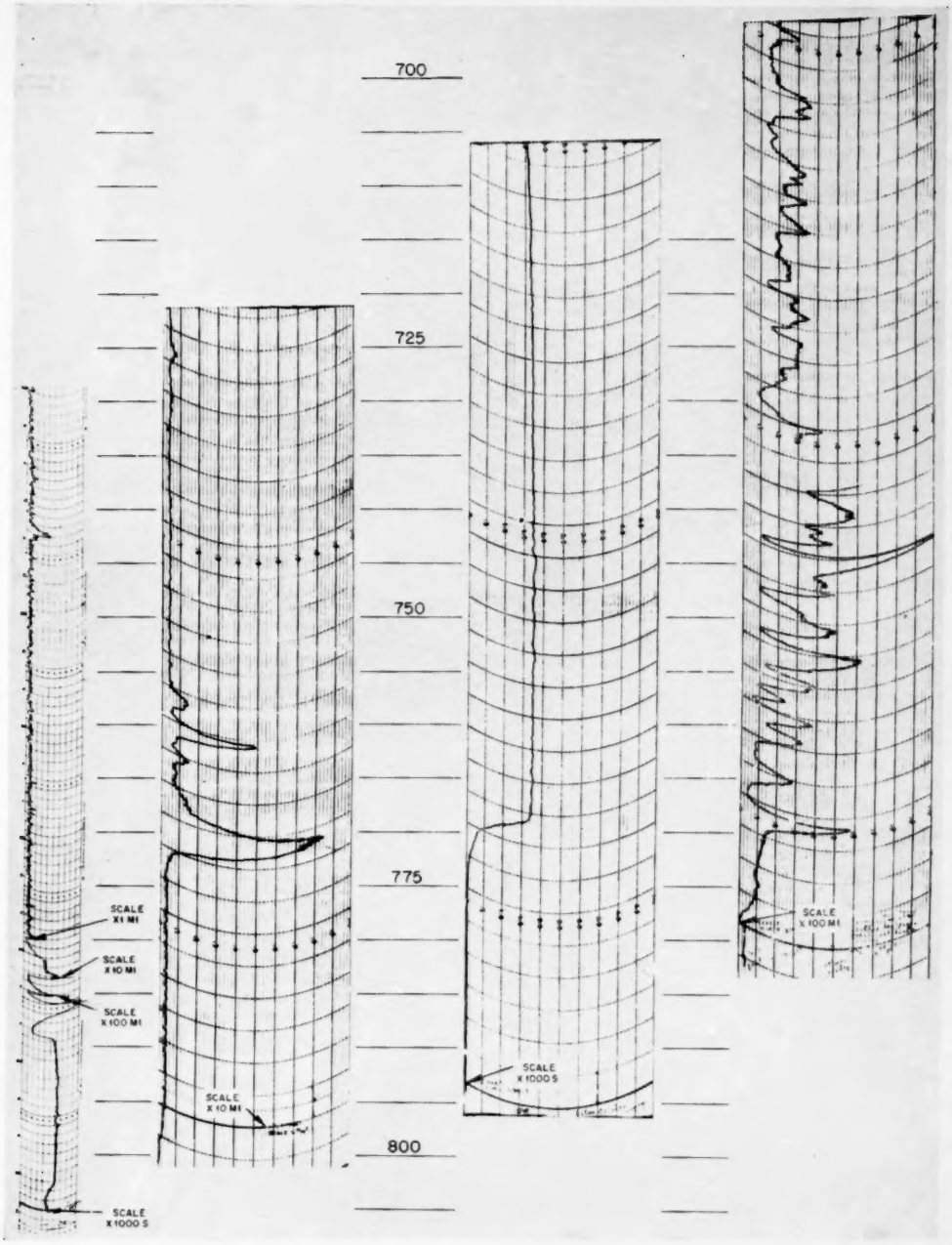


FIG. 8. Grouted well details.

detected in trees and in the litter of the immediate environs. The vehicle of transport of cesium is uncertain. It may be the ground water, or it may be airborne. Air monitoring stations were installed around the pits. The gross activity collected by standard gum paper for twenty-five weekly periods was statistically analyzed. These analyses showed that the amount of airborne activity in the pit area was significantly greater than similar off-site stations. The average concentration was about twice that of the off-site stations. Efforts are progressing to determine the specific radionuclides involved.

One undesirable feature in the use of open pits is the external radiation field. As shown in Fig. 10, radiation intensities as high as several roentgens per hour at the edge of the



(*a*) (*b*) (*c*) (*d*)

FIG. 9. Radiolog charts.—(*a*) Normal run 4 May 1956, Well No. 78. Grout Well in Pit No. 4; (*b*) normal run 9 May 1956, Well No. 82. Grout Well adjacent Pit No. 4; (*c*) normal run 17 April 1956, Well No. 74. Open boring adjacent Pit No. 4; (*d*) flush run 10 May 1956, Well No. 74. Open boring adjacent Pit No. 4.

pits will curtail the working time in the area. The intensity of the field is related to the amount of liquid in the pits since the liquid acts as a radiation shield over the side walls of the pit that have been in contact with the contaminated solution.

Future considerations. Within the extent of our present knowledge, the system of surface pits at the Laboratory has operated with apparent safety and has been useful in the disposal of large volumes of intermediate level waste. Since the system will have a finite capacity for the disposal of these wastes, suitable criteria and reclamation methods must be developed for terminating this operation. Future pits should be modified to provide an earthen shield which will eliminate any airborne contamination and external radiation.

Experimental work should be directed toward the development of pits designed to increase the extent of contact between the soil and the liquid waste. This can be accomplished in several ways: (1) by reworking the weathered shale around the pits to improve the homogeneity of the soil; (2) by locating the length of pits normal to the direction of the formation strike and thus increase the volume of shale that will come in contact with the waste; and (3) by placing a prepared soil in a lined pit and passing the waste through this filter-sorbent bed. Additional improvement will be derived by pretreating the waste to remove or reduce the objectionable anionic constituents.

Acknowledgments—The authors wish to acknowledge the assistance of H. J. Wyrick, E. R. Eastwood and R. L. Newton, in the conduct of the studies reported.

REFERENCES

1. H. H. Abee, *Problems in the Disposal of Solid Wastes at Oak Ridge National Laboratory* pp. 223–228. TID-7517 (1956).
2. J. A. Lane, *Nucleonics* **12,** 10 (1954).
3. H. R. Zietlin, E. C. Arnold and J. W. Ullman, *Nucleonics* **15,** 1 (1957).
4. F. L. Culler, *The Nature and Magnitude of Radioactive Wastes as Influenced by Types of Reactors and Fuel Processing.* ORNL CF-56-5-2 (1956).
5. C. V. Theis, *Proc. International Conference on Peaceful Uses of Atomic Energy, Geneva,* 1955. Vol. 9, Paper 564. United Nations, New York (1956).
6. A. E. Gorman, *Proc. International Conference on Peaceful Uses of Atomic Energy, Geneva,* 1955. Vol. 13, Paper 283. United Nations, New York (1956).
7. R. J. Morton, F. L. Parker, K. E. Cowser and E. G. Struxness, *An Evaluation of the Studsvik Plan, Report on the Preliminary Study of the Potentialities of Waste Disposal at the Studsvik Site.* ORNL CF-57-2-87 (1957).
8. P. B. Stockdale, *Geologic Conditions at Oak Ridge National Laboratory (X-10) Area Relevant to the Disposal of Radioactive Wastes,* AEC Technical Information Services, ORO-58 (1951).
   G. D. De Buchananne and R. M. Richardson, Ground Water-Water Resources of East Tennessee. *Tennessee Division Geology Bull.* No. 58, Part 1 (1956).
   W. de Laguna, *Some Geologic Factors that Influence Disposal of Radioactive Wastes into Pits.* TID-7517, Pt. 1b (1956).
9. T. W. Brockett. Jr. and O. R. Placak, *Proceedings Eighth Industrial Waste Conference* pp. 393–409. Purdue University, Lafayette, Ind. (1954).
   T. W. Brockett, Jr., B. Kahn and C. P. Straub, *Removal of Radioactive Cations from Solutions by Local Clay-Type Soils,* ORNL CF-54-12-145 (1954).
10. F. L. Parker and K. E. Cowser, *The Use of Soils in the Disposal of Reactor Fuel Reprocessing Wastes.* Paper presented at the American Nuclear Society Meeting, Pittsburgh, Pa. (1957).
11. *A Meteorological Survey of the Oak Ridge Area.* U. S. Weather Bureau, Oak Ridge, Tenn. ORO 99 (1953).
12. *Local Climatological Data*—1956, U. S. Weather Bureau, Oak Ridge, Tenn.
13. *Evaporation in Tennessee River Basin,* Tennessee Valley Authority. Report No. 0–5164 (1951).
14. Unpublished Information. U. S. Weather Bureau, Oak Ridge, Tenn.
15. M. A. Kohler, T. J. Nordenson and W. E. Fox, *U. S. Weather Bureau Research Paper* 38 (1955).

Health Physics Pergamon Press 1958. Vol. 1, pp. 164–168

EMERGENCY RADIATION MONITORING OF DRINKING WATER*

G. W. C. TAIT† and W. F. MERRITT
Health Physics Branch, Atomic Energy of Canada, Chalk River, Ontario, Canada

(*Received* 21 *February* 1958)

Abstract—A simple method using standard radiation survey meters is given for monitoring drinking water against bomb fallout. The γ-field associated with a suitable volume of water is measured and compared with the fields expected from permitted concentrations of fission products of known age. The fields produced by these permitted concentrations have been tabulated, and the concentrations are themselves chosen consistent with U.S. and British recommendations.

The validity of the technique is examined from a number of points of view and has been checked by experimental studies. The method is generally applicable with minor limitations whether or not water treatment has been undertaken.

INTRODUCTION

SOME time ago a simple method was developed at Chalk River for emergency monitoring of drinking water against bomb fallout using standard radiation survey meters.[1] As the hazard considered appeared slight and as British and U.S.[2,3] reports giving other techniques appeared at the same time no widespread distribution of the information was attempted. With present concern about fallout from the larger weapons, our proposed method seems to have increased value. It is less dependent on specialized equipment and of more general application than other proposed methods. The following report includes further work carried out in development of this approach.

The basis of the method is the measurement of the γ-field associated with a suitable volume of water and comparison with fields from permitted concentrations of fission products of known age.

The primary hazard from fresh bomb fallout is taken to be due to Sr^{89} and Ba^{140} ($+La^{140}$). This view parallels the U.S. and U.K. assumptions made in deriving their recommended permissible levels[3,4]. The emergency permissible intake is set at the U. K. figure of 30 μc Sr^{89} plus 150 μc Ba^{140} ingested over a 10 day period. This is expected to lead to an irradiation of 5 r to the sensitive volumes of the bones. For fresh fission products the Sr^{90} hazard is considerably less than that due to Sr^{89} and Ba^{140}. The U.S. recommendation, although not entirely comparable in form, is similar.

Two other sources of hazard exist which might conceivably be comparable in effect. They are I^{131} and the general irradiation of gut. Arguments for disregarding I^{131} as a limiting toxin are given in both the U.S. and U.K. reports. It is held that any need to hold thyroid exposures to permissible levels for whole body irradiation is not borne out by medical experience. It has also been suggested that general irradiation of the G.I. tract might be significant in the case of recently formed fission products. However, assuming a minus 1.2 power law decay and meter readings no higher than given below for tolerance doses, the maximum integrated exposure of the G.I. tract would be about 12 tissue rads of β- and γ-radiation.

The key to the method lies in the inability

* This paper is a condensation of a Chalk River Report distributed as AECL 505.

† Director, Division of Health Safety and Waste Disposal International Atomic Energy Agency, Vienna, Austria.

of natural processes or simple methods of treatment to separate the strontium and barium isotopes. That is, the proportions of Sr^{89}, Sr^{90}, Sr^{91} and Ba^{140} will be set largely by the elapsed time after formation. In addition the daughters of the above nuclides will usually be present in appropriate proportions unless unusual treatment methods have been carried out. Consequently, it is possible to estimate the concentration of toxic nuclides including pure beta emitters by measuring the γ-field from associated isotopes emitting γ-radiation.

In mixed fission products γ-radiation will be emitted by many nuclides in addition to the members of the alkaline earth family. It might be thought desirable to include the γ-radiation coming from these less dangerous nuclides in estimating the total flux from any given concentration of the toxic nuclides. Failure to include such contributions to the total flux will certainly result in an error in estimating the toxicity from the observed γ-field. However it can be shown this error will take the form of a moderate safety factor which should rarely exceed three. Further, by neglecting the contribution of nuclides which are not members of the alkaline earth family there is no need to be concerned as to whether any selective concentration of the Sr–Ba group has occurred.

The concentrations of the more toxic nuclides, Sr^{89} and Ba^{140} (plus La^{140}) are therefore estimated in terms of the radiation field from the γ-emitting isotopes Sr^{91}, Ba^{140} (plus La^{140}).

THEORY

The fallout fission products are presumed to be distributed throughout the water either in solution or suspension. In an extensive body of water in equilibrium if N is the number of quanta originating per ml per sec, then it is also the number stopping per ml/sec. This figure must also be given by the product of the flux and the absorption coefficient. At the interface between active water and inactive air the flux will be very nearly half that within the water.

$$F = N/2\mu_w \qquad (1)$$

where F is the flux in air in quanta per cm^2/sec and μ_w is the linear absorption coefficient for water.

The ionization produced in air by this flux is $F\mu_A E/w$ where μ_A is the linear absorption coefficient for air, E the energy per quantum and w the energy dissipated per ion formed.

A computed value for μ_A/μ_w combined with equation (1) gives:

$$\text{Radiation field} = 3.05 \times 10^{-8}\, NE \text{ mr/hr} \qquad (2)$$

Field from a large body of contaminated water

The activities of the various significant nuclides (relative to Sr^{89} as unity) are as in Table 1.[4]

For a 10-day consumption of 25 l, the permissible concentration is 1.2 μc Sr^{89} plus 6.0 μc of Ba^{140} per l. The corresponding concentrations of Sr^{91} (plus Y^{91}), Ba^{140} (plus La^{140}) found from Table 1 combined with γ-ray energies given in standard tables of disintegration energies are substituted in equation (2) to give the fields over a large body of contaminated water as in Table 2.

Table 1. *Relative activity of significant nuclides*

Nuclide	*Time after bomb burst*			
	12 hr	1 day	2 days	10 days
Sr^{89}	1	1	1	1
Sr^{90}	0.006	0.006	0.006	0.007
Sr^{91}	68	28	5	0
Ba^{140}	5.4	5.3	5.0	3.6
La^{140}	1.1	2.2	4.5	3.7
All fission products	1000	420	170	36

(The total γ-field for all fission products in equilibrium with permissible levels of Sr plus Ba at 1 day is about 200 mr/hr and for all the non-volatile fission products about 140 mr/hr.)

The permissible levels are set on the supposition that all the γ-radiation found arises from Sr and Ba and their immediate daughters introducing at the most, a safety factor of about three. If drastic segregation and enhancement of the most toxic components have occurred the standard of monitoring remains adequate. The

Table 2. γ-Fields at the surface of a large body of water contaminated to the maximum permissible level for 10 day consumption

	Time after bomb burst			
	12 hr	1 day	2 days	10 days
γ-Field from $Sr^{91} + Y^{91}$ (mr/hr)	98	40	7.2	0
γ-Field from Ba^{140}	2.6	2.4	2.3	1.7
γ-Field from La^{140}	3.4	6.6	13.3	11.2
Total γ-field Sr + Ba (mr/hr)	104	49	22.8	12.9

only necessary precaution is in connection with the presence of La^{140} which is the major contributor to the γ-field after 2 days. Some forms of water treatment might possibly remove La relative to its Ba precursor. Therefore, complete assurance as to water safety would require that in testing water more than 2 days after the bomb functioned, a day elapse between any drastic water treatment and final monitoring. However, it is doubted if any water treatments likely to occur in practice will separate Ba and La.

An additional advantage in using the monitoring levels set on the assumption that Sr (+Y) and Ba (+La) are the sole sources of γ-radiation lies in the fact that such levels do not raise any question of gastrointestinal damage from general irradiation. If the levels were raised appreciably a dose to the gut might be achieved which some authorities regard as objectionable.

The recommended technique is to hold a γ-indicating survey meter close to the surface of any large body of water. Meter readings are then compared with the values in the last row of Table 2 for the appropriate time after the bomb exploded.

With longer periods of consumption, the initial concentration need not be reduced proportionately since much of the Ba^{140} activity will have decayed prior to ingestion.

The γ-fields corresponding to the permissible concentration for 30 day ingestion are found to be one half those given above for 10 day ingestion.

Fields from limited volumes of water

The range of application of this technique is increased if it can be applied to typical water containers. The following water containers were studied: (a) 1 l. aluminum canteen; (b) 2 gal (Canadian) light steel can: 12 in. × 8½ in. × 5½ in.; (c) 4 gal (Canadian) heavy steel can: 15 in. × 13 in. × 6 in., "Jerry-can".

Fields to be expected for these and other vessels when filled with water contaminated to the maximum permissible level were calculated in the following way. The container was considered replaced by a cylindrical vessel with its section equal in area to the cross-section of the vessel perpendicular to its long axis. The γ-flux at the surface of such a cylinder of infinite extent was then calculated using a suitably modified form of equation (2):

$$\text{Radiation field} = 3.05 \times 10^{-8} \left[2 + 2r\mu_w - 2r^2\mu_w^2 \left\{ I_0(r\mu_w)K_0(r\mu_w) + I_1(r\mu_w)K_1(r\mu_w) + \frac{I_0(r\mu_w)K_1(r\mu_w)}{r\mu_w} \right\}\right] NE \text{ mr/hr} \quad (3)$$

where r is the radius of the cylinder, μ_w the absorption coefficient neglecting Compton scattering and K_0, K_1, I_0 and I_1 are the usual Bessel functions. The correction for the finite length of the container was considered in the following way. The field was found for an infinite rod lying along the axis of the cylinder loaded with radioactivity per unit length equal to that per unit length of the cylinder. The correction factor for a rod of finite length was calculated and this factor applied to the values given by equation (3). This approximation was considered reasonable since the end correction is not great and the rod and cylinder calculations give approximately the same fields if the radiation mean free path is greater than the radius of the cylinder. For larger tanks the correction

is probably slightly overestimated but this will set an upper limit to the estimated concentration and provide a safety factor. Table 3 can then be calculated.

Table 3. γ-Radiation field from permissible levels of fresh fission products in drinking water (mr/hr)

(Fields are to be measured at the center of one surface and correspond to the ingestion of 30 μc Sr^{89} plus 150 μc Ba^{140} over a 10day period.)

Object	*Time after bomb burst*			
	12 hr	1 day	2 days	10 days
Large body of water	104	49	23	13
Tank truck (1000 gal.)	68	32	15	8.5
4 gal Jerry can	28	13	6	3.5
2 gal oil can	25	12	5.5	3.
Water bottle (1 l.)	13	6	3	1.5

Experimental verification

The validity of equation (2) was originally tested[1] with Na^{24} dispersed in a large tank of water.

The accuracy of the estimation of the fields from the smaller irregular bodies of water was now investigated using material more closely approximating the toxic fallout. A solution of 55 μc/l. of Ba^{140} was used after reaching equilibrium with its La^{140} daughter. Measurements were also made with a solution of 11 μc Ba^{140} (plus La^{140})/l. The water containers tested were the 1 l. bottle and the 2 and 4 gal cans. γ-fields were measured using two types of survey meters, an ion-chamber type (Tracerlab–SU–1F) and a halogen geiger type (Atomic Energy of Canada 1922). The latter is designed to read directly in mr/hr. Each survey meter was placed in contact with the center of the broadest face of each vessel. Table 4 compares the measured fields with calculated values found as for Table 3.

Agreement is good except in the case of the 1 l. canteen. The low values for the measured fields in this case are obviously due to the fact that the size of the sensing element of the meter is large compared to the dimensions of the vessel and thereby fails to give a reliable measure of the γ-field at the vessel's surface. This is borne out by the observation that the AEP 1922 instrument with smaller chamber gives more nearly the calculated field.

Materials likely to be used in construction of tanks and water trucks were tested by insertion between the sample vessel and the meter: $\frac{1}{8}$ in. aluminum and $\frac{1}{8}$ in. and $\frac{1}{4}$ in. bakelite had no measurable effect. The plastic was considered equivalent to thermal lagging materials likely to be used in practice.

Practical application

The calculations and observations made in the previous sections can be applied very simply. The chief modification required is to round off some of the figures and generalize requirements. Table 5 is the basic table proposed.

The above table applies whether the water has been subject to treatment or not. The only exception would be for fallout more than 2 days

Table 4. γ-Field at the surface of containers

Vessel	Concentration of Ba^{140} (μc/l.)	Observed field		Calculated field (mr/hr)
		SUIF (mr/hr)	AEP 1922 (mr/r)	
4 gal can	55	40	40	44
	11	8	8	8.7
2 gal can	55	32	32	39
	11	6.5	—	7.8
1 l. can	55	8.5	13.5	20

Table 5. γ-Field measurements (mr/hr) *for maximum permissible water contamination*

This table applies to any fresh fallout contamination and is for 10 day consumption. Values should be halved for 30 day consumption.

Water body	*Time since bomb burst*			
	12 hr	1 day	2 days	10 days
Reservoir or lake, measured far from shore	100	50	25	12
Reservoir, pond, etc., measured at arms length from shore, close to surface and over water at least 2 ft. deep	50	25	12	6
Water tank, from 150 to 1000 gal measured in contact with center of one surface	50	25	12	6
Water can, from 2 to 4 gal	25	12	6	3

old and subject to unusual chemical treatment. To cover all possibilities in such a case it might be desirable to allow a day to elapse between treatment and final monitoring. Water that has been contaminated with old fission products cannot be monitored by this method and recourse would be necessary to full radiochemical analysis. If more than 2 weeks elapse between the contaminating incident and monitoring, radiochemical analysis is suggested. However, this is not expected to arise as a practical military or civil defense problem.

The monitoring of individual containers of capacity less than 2 gal is not recommended because of the large effects of chamber size on response. If necessary, however, the user could easily compare fields from small and larger containers holding samples from the same contaminated water supply. In this way, permissible fields could easily be determined for any small containers and meters available.

REFERENCES

1. G. W. C. Tait, *Monitoring of Drinking Water for Fresh Fission Product Contamination.* CR-RHC-509 (1952).
2. W. R. Loosemore, *Emergency Monitoring of Water Contaminated with Fission Products.* AERE El/R-1046 (1952).
3. J. B. Hursh, S. Zizzo and A. H. Dahl, *Use of Commercially Available Portable Survey Meters for Emergency Fission Product Monitoring of Water Supplies.* UR-180 (1951).
4. A. C. Chamberlain, *Emergency Tolerance for Fission Products in Drinking Water.* AERE HP/M-23 (1951).

Health Physics Pergamon Press 1958. Vol. 1., pp. 169–175

IMPROVEMENTS IN THE THRESHOLD DETECTOR METHOD OF FAST NEUTRON DOSIMETRY

P. W. REINHARDT and F. J. DAVIS

Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee,
Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission

(*Received* 13 *March* 1958)

Abstract—Improvements have been made in the threshold detector method for measuring fast neutron spectra and tissue dose. The sensitivity of the counters used to measure the induced activity of the detecting elements has been increased by a factor of 12 to 16 times by improving counting geometry. Holding the gain of the scintillation counting system constant by an automatic monitoring arrangement reduces the time required for standardization. The method of P^{32} recovery from S^{32} by burning out the sulfur has been shown to be reproducible and to have a recovery efficiency of 93.5 per cent. The neutron cross-sections for various detection elements used in the threshold detection system are reviewed.

1. INTRODUCTION

THE method of measuring the spectrum of fast neutrons with a series of foil detectors has previously been described by HURST *et al.*[1] The detectors consist of Au and Au plus Cd for determining the thermal flux, Pu^{239} shielded with B^{10} for determining the total fast flux, Np^{237} for determining the flux above 0.75 MeV, U^{238} for determining the flux above 1.5 MeV, and S^{32} for determining the flux above 2.5 MeV. The amount of activity induced by fast neutrons in the detecting elements is determined with scintillation counters which are calibrated by irradiation of suitable foils with thermal neutrons. From the measured spectra, the dose can be determined by:

$$\begin{aligned} D &= [0.028\ N_T + 1.0(N_{PU} - N_{NP}) \\ &\quad + 2.4(N_{NP} - N_U) + 3.0(N_U - N_S) \\ &\quad + 3.7\ N_S] \times 10^{-9} \\ &= [0.028\ N_T + 1.0\ N_{PU} + 1.4\ N_{NP} \\ &\quad + 0.6\ N_U + 0.7\ N_S] \times 10^{-9} \end{aligned} \qquad (1)$$

where D is the tissue dose (rad), N_T is the number of thermal neutrons per cm^2, and N_{PU}, N_{NP}, N_U, and N_S are the number of n/cm^2 above the thresholds for Pu, Np, U, and S, respectively.

As a result of the increased need for measuring neutron spectra for dose determinations in the event of reactor accidents, pulses of intense neutron flux, and other applications, several improved techniques have been developed. Most of these improvements have been aimed at increased sensitivity and ease of handling of large numbers of foils.

2. INSTRUMENTATION

Increased sensitivity could be obtained if the detector samples were increased in size. However, since the world's supply of Np^{237} is definitely limited, it is more feasible to increase the sensitivity by increasing the counting efficiency.

In the original method described by HURST *et al.*[1] the fission foils were counted on a 1 in. × 1½ in. NaI crystal γ-counter. In order to improve the sensitivity the crystal size was increased to 4 in. diameter by 2 in. thick and two crystals were employed, one on each side of the foil (Fig. 1) the ⅛ in. brass filter was replaced by a ⅝ in. lead filter. The increased filtering is especially effective in reducing the natural soft-γ-background of Np^{237}.

The discriminator was set so as to accept all pulses of a height greater than those due to the

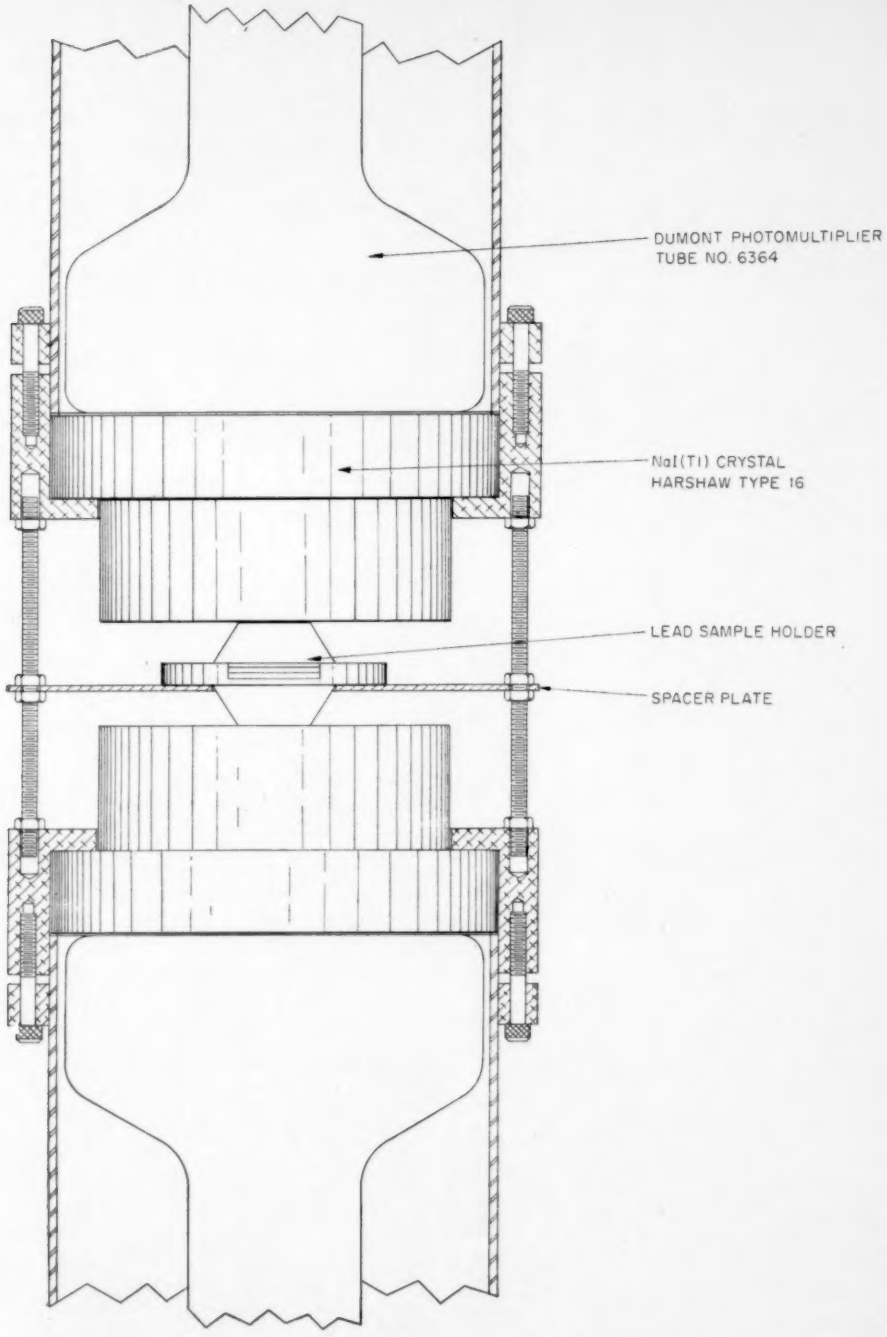


FIG. 1. Fission foil scintillation counter.

total absorptions of 0.66 MeV γ-rays. Among other factors involved in this choice of discriminator setting was the convenience of using the 0.66 MeV γ-ray of Cs^{137} as a monitor. The over-all gain in count rate over the previous system was a factor of 12.4 at 1 hr after irradiation and 15.6 at 12 hr after irradiation of the fission foils.

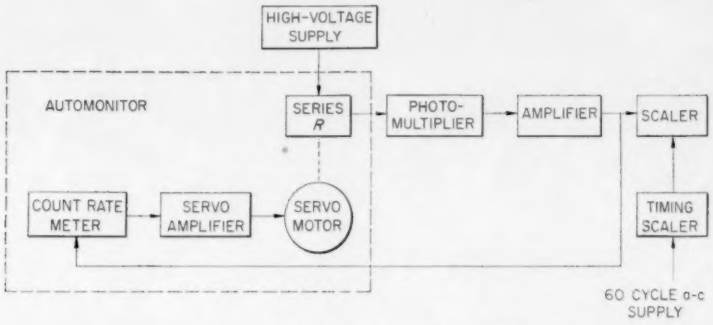


FIG. 2. Block diagram showing automatic monitoring system.

An automatic monitoring system (Fig. 2) was developed to reduce error due to drifts in gain in the scintillation counting system. In this system a Cs^{137} source is automatically placed in the foil-counting position each time a sample foil is changed. This activates a servomotor which adjusts the gain of the system so that the count rate is standardized. Since the count rate of the Cs^{137} source at the discriminator setting of 0.66 MeV is particularly sensitive to gain, the servo has only to maintain this count rate to within 10 per cent in order to maintain the stability of count rate of the fission foils to within 1 per cent. Ordinarily the servo system can maintain the count rate of the Cs^{137} to within 1 or 2 per cent.

In the older technique of counting, an automatic timer was used to cut off the count at a predetermined time. This timer utilized a synchronous motor with clutches and relays. As a result of slippage of clutches, etc., the method was found to be unreliable in short counting times of the order of 1 min; therefore, a timing scaler was substituted for the synchronous motor. The timing scaler counts the cycles of the 60 c a.c. supply and, when set to a predetermined count, cuts off the counting scaler so that it counts over a time interval accurate to within 1 c. Thus it has been found convenient to refer to count rates as counts/kc rather than counts/min. One kc equals $16\frac{2}{3}$ sec.

In order to correct for the decay of the fission-foil γ-count rate with time, it has been found advantageous to record the correction factors at 1 min intervals on an Esterline–Angus (EA) recorder tape. The tape is then run through an EA recorder and the correction factors read off while the foils are counted. This eliminates the necessity for recording the time elapsed since irradiation and the subsequent conversion of this time to a decay factor.

3. FISSION FOIL SYSTEM

Previously, when Np^{237} fission foils were being counted, it appeared that the count rate did not decay at the same rate as the count rate for Pu^{239} fission foils during the first several hours after irradiation. One reason is that Np^{238} (produced by neutron capture in Np^{237}) which decays with a 2.1 day half-life, emits γ-rays of the order of 1.0 MeV. By counting the Np foils at approximately 2 days after irradiation, the amount of activation due to neutron capture can easily be calculated. If the Np^{237} foils are counted at a discriminator setting equivalent to 1.2 MeV, the 1.0 MeV γ-ray from Np^{238} is eliminated. This reduces the fission γ-ray counting rate by about a factor of 3 and reduces the Np^{237} background to a negligible amount compared to the background of the counting system. In cases where the activation count rate was of the order of 20 per cent of the total count rate the use of the higher discriminator setting gave better data.

It was further noticed that the U^{238} foils did not decay as fast as Pu^{239} but did decay faster than Np^{237}, suggesting that a shift in the yield of the fission products may be the phenomenon being observed. In order to test this a U^{235} foil was irradiated and its decay curve followed. It was found that the difference in the count rates of U^{235} and Pu^{239} at any particular time after fission was twice the difference in the count rates of Np^{237} and Pu^{239}. This indicates that the change in the yield of the fission products was proportional to the mass of the parent atom. The decay curves of the fission product γ-rays of Pu^{239}, U^{235}, and Np^{237} as measured by our counting system are shown in Fig. 3. It is noted that the Pu^{239} and U^{235} curves at first diverge, reaching a maximum separation at approximately 5 hr and then converge at about 30 hr. In studying the Hunter–Ballou[2] curves for the fission products most prominent at 5 hr after irradiation and which emit γ-rays more energetic than 0.66 MeV, and assuming that the curve[3] giving fission product yield vs. mass number would shift in the direction of increasing mass number proportional to the mass of the parent atom, it appears that Te^{133}, Y^{92}, I^{135}, Rb^{88} and Kr^{87} would increase their relative yield in going from Pu^{239} to U^{235}, while only La^{141} would decrease its relative yield. The yield of Cs^{138} would probably remain about the same since it was on the peak of the yield curve. The arguments noted above indicate changes in the decay curves in qualitative agreement with experiment.

Assuming that the decay curves for Np^{237} and U^{238} deviate from the Pu^{239} curve (Fig. 3) by the ratio of the difference in their atomic weights, Np^{237} should deviate one-half as much as U^{235} and U^{238} by one-fourth. The curve for Np^{237} (shown in Fig. 3) is for a higher discriminator setting (1.2 MeV) and the count rate is reduced accordingly.

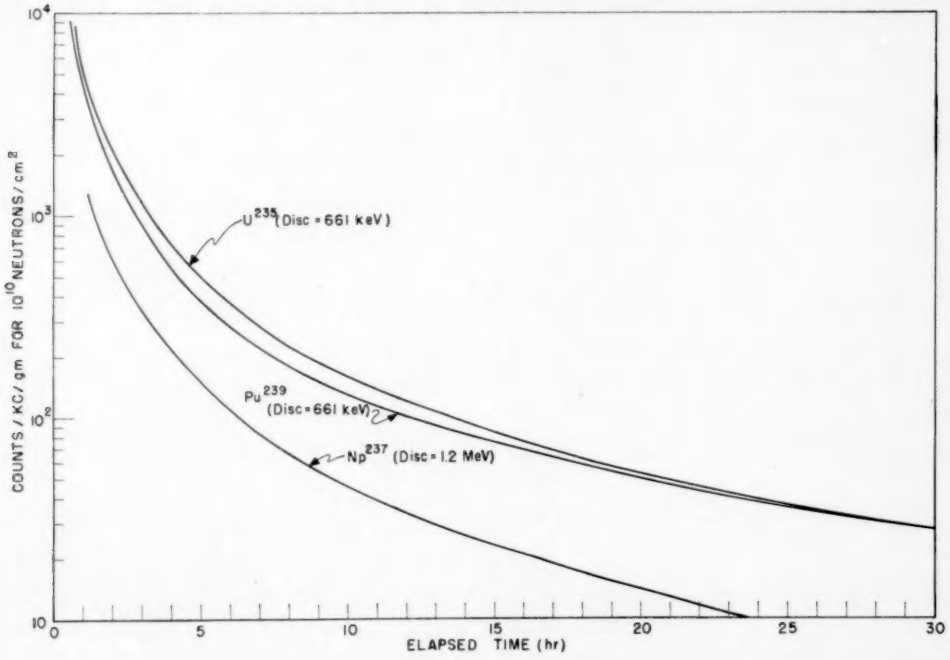


FIG. 3. Fission product decay curves.

4. SULFUR COUNTING

Until recently, accuracy of determination of the number of neutrons having energies greater than 2.5 MeV (by counting the 1.71 MeV β-rays from the $S^{32}(n,p)P^{32}$ reaction) has been limited

by the area of the sulfur sample. Although larger sulfur samples could be used, larger counters would be required, and counter backgrounds would be increased accordingly. A simple method for removing the P^{32} from the sulfur has been developed, making possible manyfold increase in the sensitivity of a given detection system with no increase in counter background.[4]

Phosphorus-32 is removed from sulfur by first melting the exposed sulfur sample in an aluminum dish; a hot plate operated at a low temperature is adequate. The sulfur is then ignited and allowed to burn out leaving the P^{32} attached to the aluminum dish. It has been found that very pure sulfur is required for a complete sulfur burn out.

The sulfur is burned in a 0.001 in. thick aluminum dish $1\frac{1}{2}$ in dia. and $\frac{1}{2}$ in. deep. After burning, the sides of the dish are folded down forming a $1\frac{1}{2}$ in. disk which is counted on the scintillation counter shown in Fig. 4. For increased counter geometry, a lead reflector 1 in. dia. and $\frac{1}{8}$ in. thick is placed on top of the

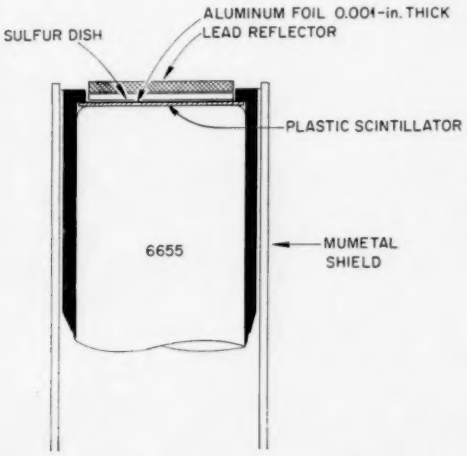


FIG. 4. Scintillation counter assembly for counting P^{32}.

aluminum disk. The counting geometry of the system was found to be 61 per cent, when the counting bias was chosen to be just sufficient to exclude photomultiplier noise. A natural-uranium source is used to standardize the counter. A standard made of six pieces of U $\frac{1}{8}$ in. in diameter by $\frac{1}{8}$ in. thick gives 4370 count/kc.

A burned out sulfur sample should give the same count rate as an unburned sample of zero thickness if all of the P^{32} is recovered. In Fig. 5 is shown a plot of counts/min per g vs. weight (g)

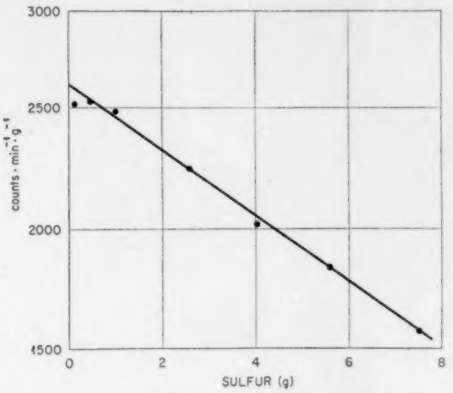


FIG. 5. Extrapolated activity of P^{32} in sulfur.

of sulfur. The data were obtained by alternately increasing the amount of sulfur sample and counting. In each case the sulfur was weighed and spread uniformly over the counting surface of the detector. The extrapolated zero thickness count rate (Fig. 5) is 2605 counts/min per g. The entire amount of sulfur used in this extrapolation was then burned out and a count rate was 2395 counts/min per g. From these count rates a P^{32} yield of 92 per cent is obtained. For further verification of the P^{32} yield, a quantity of P^{32} was spread over the surface of a standard sulfur pellet, and the pellet placed with the active side face down in an aluminum dish and counted. The sulfur was then turned active side up and melted and burned as previously described. The sides of the dish were folded, and the dish placed on the counter. To maintain the counting geometry of the activated pellet, an unactivated pellet was placed on top of the dish. The yield determined by this method was 93.5 per cent. This method is considered more accurate since both counts are taken with identical geometry and higher counting rates could be used to reduce the statistical error. Applying the 61 per cent counting geometry to the 93.5 per cent yield, one obtains a geometry

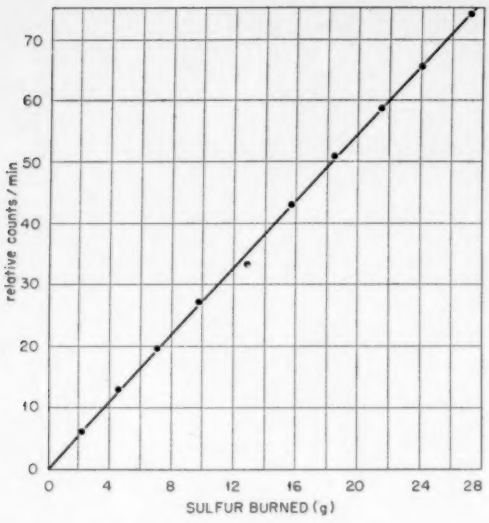


FIG. 6. Count rate after various amounts of sulfur were burned.

of 57 per cent of the total P^{32} disintegrations for a given burned out sulfur sample. To determine the reproducibility of the method, ten consecutive burnings were made in the same aluminum dish and from a single batch of neutron-irradiated sulfur. The count rates as determined after each burning are shown in Fig. 6. The point at 12.9 g is undoubtedly a counter error since the buildup of activity falls back on the curve after this point. An exposure of the standard 21 g ($1\frac{1}{2}$ in. by $\frac{3}{8}$ in.) pellet to 10^{10} n/cm^2 above the 2.5 MeV threshold gives a count rate of 235 counts/kc. When burned, the count rate is 4370 counts/kc or 210 counts/kc per g. A recent exposure at the Tower Shielding Facility gave a count rate of 58 counts/kc for a 21 g burned pellet or 2.8 counts/kc per g for a total neutron dose of 1.0 rad. These counting rates are to be compared to the background of 8 counts/kc for the scintillation detector.

5. CALIBRATION CROSS-SECTIONS

Since publication of the threshold detector article,[1] better determinations have been made of the neutron cross-sections of the various detectors. All the cross-sections discussed here are values given in BNL–325.[5] The fission cross-section for Pu^{239} is 720 barns at thermal energy[6] but at room temperature the non-1/v correction is 1.075; thus the adopted cross-section is 774 barns. The $S^{32}(n,p)P^{32}$ reaction is calibrated by activation of P^{31} with thermal neutrons; the cross-section for the latter is 0.19 ± 0.03 barns. Note that BNL–325[5] recommends the absorption cross-section as being more reliable for this case.

The adopted fission cross-sections for Pu, U, and Np in the fast neutron region remains the same as those previously published.[3] The cross-sections of the $S^{32}(n,p)P^{32}$ used are "smoothed out" values taken from ALLEN *et al.*[7] Since the cross-section changes with energy, it was weighted by the fission spectrum[8] to determine an effective value. This effective value depends on the adopted threshold in the following way:

$$\sigma_{\text{eff}} = \frac{\int_0^\infty n(E)\sigma(E)\,dE}{\int_{E'}^\infty n(E)\,dE} \qquad (2)$$

where $n(E)$ is the spectral distribution function, $\sigma(E)$ is the cross-section for the $S^{32}(n,p)P^{32}$ reaction, and E' is the threshold energy. When $E' = 2.0$ MeV, σ_{eff} is 0.173 barns; and when $E' = 2.5$ MeV, σ_{eff} is 0.229 barns. We have chosen to use the latter values ($E' = 2.5$ MeV and $\sigma_{\text{eff}} = 0.229$ barns) in measuring fission type spectra.

6. CALIBRATION MEASUREMENTS

The entire threshold detector system was calibrated by irradiating Au, Pu^{239} and P^{31} with thermal neutrons. Gold samples were exposed both at the Los Alamos Water Boiler and in the water column at the top of the ORNL graphite reactor. It was found that the fluxes in the two reactors were in good agreement with each other; however, the ORNL value had been determined by comparison with the flux in the X-10 Standard Graphite Pile.[9] The flux in the X-10 Standard Pile is an "absolute flux" which should be converted to an "activation flux". When this is done, the flux values in the Los Alamos Water Boiler and the X-10 Standard Pile are reduced by the factor 1/1.13.

In making a calibration with thermal neutrons

Table 1. *Ratio of threshold detector value to* ROSEN's *value*

Element	Run 1	Run 2
Pu	1.00	1.00
Np	1.23	1.14
U	1.11	0.91
S	0.95	0.95

it is also necessary to correct for flux depression. The "depression" in flux was measured by thin (1 mil) gold foils with and without the 0.1 g Pu^{239} calibration foil between them. The measured factor was 0.82. Calculations using SKYRME's theory[10] gave the factor as 0.821 and showed that the flux correction when averaged through the sample was 0.80. At the water boiler the nominal integrated flux for the Pu^{239} calibration exposure was 9.6×10^8 n/cm, and this was corrected to an effective value of $(9.6 \times 10^8 \times 0.80)/1.13 = 6.8 \times 10^8$ n/cm.

Comparison measurements of neutron dose were made at the ORNL Tower Shielding Facility using the threshold detector system and the Hurst proportional counter.[11] The measured doses were 54.2 and 53.8 rads, respectively. Spectrum measurements made at the LASL Godiva facilities were compared to values given by ROSEN.[12] The values normalized for Pu^{239} are given in Table 1. Reasonable agreement is seen except for the Np^{237} flux. Perhaps this error could be explained as an error in the Np cross-section.

Table 2 gives the relative sensitivity and background counts of the foils. The counting system for the Au is the same 1 in. $\times$ 1½ in. NaI crystal arrangement as described by HURST *et al.*[1]

REFERENCES

1. G. S. HURST, J. A. HARTER, P. N. HENSLEY, W. A. MILLS, M. SLATER and P. W. REINHARDT, *Rev. Sci. Instrum.* **27,** 153 (1956).
2. H. F. HUNTER and N. E. BALLOU, *Nucleonics* **9,** C-4 (1951).
3. The Plutonium Project, *Rev. Mod. Phys.* **18,** 539 (1946).
4. F. J. DAVIS, J. A. AUXIER, G. S. HURST, J. A. HARTER, E. B. WAGNER and T. D. STRICKLER, *Health Physics Semiann. Prog. Rep.* p. 93. ORNL–2151 (1956).
5. D. J. HUGHES and J. A. HARVEY, *Neutron Cross Sections.* BNL–325 (1955).
6. D. J. HUGHES and J. A. HARVEY, *Heavy Element Cross Sections* (Presented at Geneva August 1955), Addendum to BNL–325 (1955).
7. L. ALLEN, JR., W. A. BIGGERS, R. J. PRESTWOOD and R. K. SMITH, *Phys. Rev.* **107,** 1363 (1957).
8. L. CRANBERG, G. FRYE, N. NERESON and L. ROSEN, *Phys. Rev.* **103,** 662 (1956).
9. E. D. KLEMA, R. H. RITCHIE and G. MCCAMMON, *AECD*–3590 (1952).
10. T. H. R. SKYRME, *Reduction in Neutron Density Caused by Absorbing Disk.* Report MS–91.
11. G. S. HURST, *Brit. J. Radiol.* **27,** 353 (1954).
12. L. ROSEN, *Proceedings of the International Conference on Peaceful Uses of Atomic Energy Geneva,* 1955. Vol. 4, p. 97. United Nations (1956).

Table 2. *Relative sensitivity and background counts of various foils*

Element	Discriminating setting (MeV)	Total background (counts/kc per g)*	Counts/kc per g for 10^{10} n/cm† at 1 hr after exposure	Counts/kc per g for 10 rads Godiva neutrons at 1 hr after exposure
Au	0.33	14	9360	
Pu	0.66	900	4390	1600
Np	0.66	700	3300	1100
Np	1.20	100	1240	400
U	0.66	400	1190	200
S "burned"		8	210	15

* 1 kc = 16⅔ sec

† Above the foil threshold

Health Physics Pergamon Press 1958. Vol. 1, pp. 176–183

POSSIBLE HEALTH HAZARDS ASSOCIATED WITH THE LAUNDERING OF RADIOACTIVELY CONTAMINATED PROTECTIVE CLOTHING*

S. G. PEARSALL, L. GEMMELL
Health Physics Division, Brookhaven National Laboratory

A. BRESLIN
New York Operations Office, U.S. Atomic Energy Commission

(*Received* 5 *December* 1957)

Abstract—An engineering study was made at the Brookhaven National Laboratory to evaluate the possible health hazards to laundry personnel and the public connected with the operation of a decontamination laundry. Laboratory clothing was separated into eight groups, ranging from only slight amounts to rather large amounts of contamination. The groups contained from nineteen to thirty-one coveralls each, with all of the garments in the same group containing about the same amount of radioactive contamination. Each group was washed as a separate batch. Air samples were taken at various breathing zones and at various times during the operation to determine the airborne dust hazard. The waste water was assayed for activity, the dryer effluent was monitored and the equipment, building, etc., were checked for contamination during the laundry operation of the eight groups.

The data show that the greatest hazard lies in the inhalation of airborne radioactive dusts. Washwater, laundry equipment, surfaces and floors became contaminated. Clean clothes showed cross contamination if washed with even slightly contaminated clothing. The dryer effluent was invariably contaminated with radioactive lint. It was evident that below 1–2 μc of contamination per garment, one could send radioactive clothing to a commercial laundry without hazard to the laundry operators or to the public. Higher levels can be handled safely by a commercial laundry if the loading and sorting operations are carefully managed.

THE Brookhaven National Laboratory, like most other major AEC installations, provides its workers with protective garments to be worn in areas where radioactive contamination is present. Since several hundred items of protective clothing per week are either contaminated or potentially contaminated, a substantial laundry facility has been needed to clean these garments for re-use. Such a decontamination laundry, run entirely by laboratory personnel, has been operating on the site for the past 6 years.

As more and more industrial groups get into the nuclear field, they are faced with the problem of laundering company owned protective clothing. Uncertainty exists as to whether it is necessary to establish their own laundries or whether they may be able to send their contaminated clothing to commercial laundries. In other words, just what are the limiting factors or special requirements of a decontamination laundry?

The problem

Specifically, the over-all health problem can be reduced to at least six questions for which answers were sought. Each of these must in turn be evaluated at different levels of radioactivity before a satisfactory answer can be made.

(1) What is the hazard from both whole

* Work performed under the auspices of the U.S. Atomic Energy Commission.

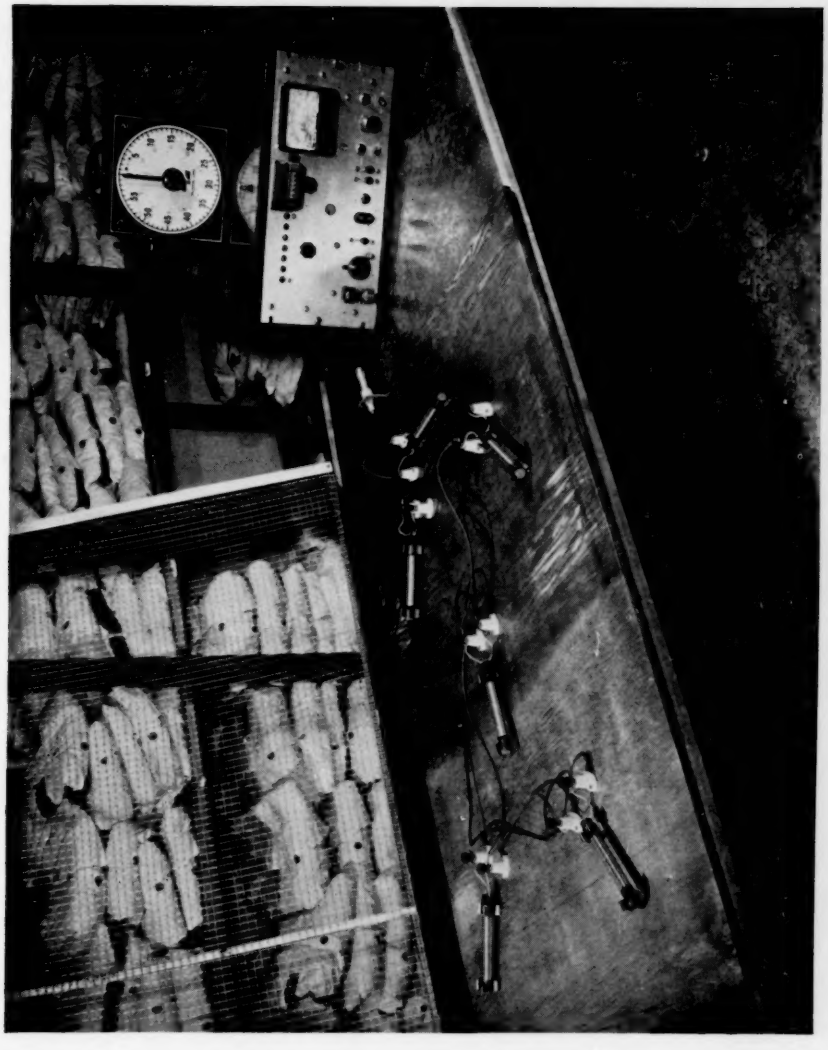


FIG. 1.

body exposure and airborne inhalation, to the persons actually doing the laundry work?

(2) What amounts of activity are released to the environment, both to the atmosphere and in the liquid waste effluent?

(3) To what extent are floors and equipment contaminated in the operation?

(4) What is the effect upon clean garments washed in a machine with contaminated garments?

(5) What is the effect upon clean garments when washed in a machine which has previously washed contaminated garments?

(6) Can garments be sorted with normal radiation survey instruments and techniques so as to minimize or eliminate any possible hazards?

The laundry facility

The laundry in which this work was done contains no special equipment for radioactive decontamination. The washing, drying and pressing equipment is located in a converted army mess hall which has no special features other than those found in any commercial laundry, with the exception that one washing machine drains to hold-up tanks. This hold-up system, with its proportional sampler, provides a fairly accurate means of assaying the wash water for radioactivity before it is released to the sewer. The washer, extractor, dryer and presser are all standard commercial machines similar to those found in any laundry establishment.

Radiation monitoring equipment

One special monitoring device was constructed for this study. It consists of a large flat box about 3 ft wide and 5½ ft long with sides 6 in. high. Six thin wall beta–gamma counter GM tubes are mounted on the bottom in parallel and in such a configuration as to monitor the radioactive contamination on a pair of coveralls (see Fig. 1). A hardware cloth screen was stretched over the box frame and supported the garments in a fixed position at a distance of 6 in. from the GM tubes. Background for the machine averaged about 900 counts/min.

In order to calibrate this multiple counter, fifteen garments were contaminated with a known amount of slurry solution of old mixed fission products obtained from the liquid waste evaporation plant. The activity of this solution was found to be about 1.3 μc/ml. From 1 to 18 ml were pipetted onto the front surfaces of new coveralls. On some of the garments the activity was uniformly spread, while on others equal amounts of activity were applied to localized areas. There appeared to be no significant difference between the counting rates for the two methods of applying the contamination. Fig. 2 represents the curve obtained when microcuries of radioactivity applied was plotted against net counts/min.

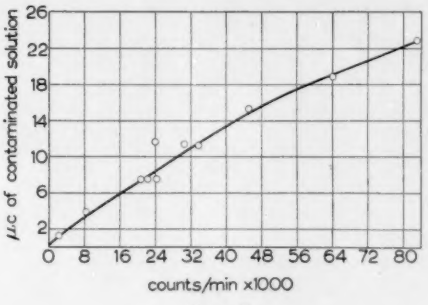


FIG. 2.

Air sampling was done at the rate of 20–30 l/min with small portable electrically operated samplers. The samples were collected on Whatman 41 filter paper 3 cm in diameter and counted in the Health Physics counting room, using end window GM counters.

Garments

The garments used for this study were cotton twill coveralls such as are in common use in the radiation areas throughout the laboratory. All garments used at the reactor are routinely monitored with a portable geiger counter survey instrument after use and separated into two groups before return to the laundry—those reading less than and those reading more than 2 mreps/hr (radium calibration). These two groups are routinely washed separately to minimize cross contamination. After the coveralls are washed, they are again scanned with a geiger survey instrument and, if not showing greater than an arbitrary acceptable level of contamination, are returned to the shelves for re-use. In general, most contaminated garments, even after being washed, do show some residual contamination. In many cases this residual contamination is not readily detectable by using an ordinary GM survey instrument. It is detected, however, in almost all cases by the counting equipment previously described and used for this study.

Inasmuch as a large part of the data to be obtained from this work was to be obtained from air samples, it was felt that the so called "fixed" contamination in the garment was not a fair estimate of the potential airborne hazard which existed. Therefore, an

Table 1. *Contamination levels of garments laundered* (counts/min per garment)

Test group No.	No. of garments washed	Range of loose contamination	Average loose contamination
1	25	0–2500	575
2	27	0–2500	710
3	21	2500–7000	3160
4	31	7000–13,000	8650
5	27	13,000–23,000	16,300
6	21	23,000–40,000	32,300
7	23	40,000–160,000	77,000
8	19	40,000–160,000	73,300

attempt was made to distinguish between that which would not readily wash out (fixed) and that contamination which was put on the garment in its most recent wearing (loose). This was done by monitoring the garment before it went back to use and again as soon as it was returned to the laundry after being worn.

Approximately three hundred and fifty garments were used in this survey. Of these, two hundred and fifty were already in routine use at the start of the study and one hundred new garments were put into use during the tests. Each garment was monitored before and after use and was classified into one of the categories shown in Table 1 prior to washing.

Each of these groups was washed before routine daily work commenced in the laundry and each on a different day to avoid cross contamination of air samples. The wash cycle used for these tests was similar to that used by commercial laundries and was as follows:

(1) Break 140°F—5 in. of water, 5 min (commercial detergent).

(2) Break 160°F—5 in. of water, 10 min (commercial detergent).

(3) Bleach 150°F—5 in. of water, 10 min (2 qt 1 per cent bleach).

(4) Rinse 160°F—10 in. water, 5 min.

(5) Repeat 4.

(6) Rinse—split $\frac{160°F}{140°F}$ 10 in. water, 5 min each.

(7) Rinse 90°—10 in. water, 5 min.

(8) Sour 4 in. of cold water ($1\frac{1}{4}$ oz sour), 5 min.

(9) Rinse with cold water (60°–70°F) to 10 in. water and add bluing; run 5 min.

With this background, we are ready to consider the original questions and present the data which was obtained in answer to them. It must be kept in mind that in each test case we are considering one wash only and thus have not considered the additive effect of additional washes.

Exposure Hazards

A check of the personnel monitoring files of the two men who handle all pick-up and delivery as well as the washing of all contaminated clothing, showed negligible exposures from 1 July to 31 December 1956. Routine bioassays of urine specimens of these two men have failed to show any appreciable body burden of radioactivity. However, as a precautionary measure they have been required to wear a respirator with high filtering efficiency at all times when loading a machine or sorting contaminated clothing so that no urinary excretion is to be expected.

Waste effluents

During the tests reported here, air samples were taken both for the breathing zone and for the general room air at the center of the room, for each stage of the laundry operation. The results of these tests are given in Tables 2 and 3.

Since there were no data as to how long the operators' breathing zone hazard persisted, it was decided to wash four contaminated batches of clothing and to sample air at different times after the maximum hazard condition was created. Table 4 shows the decrease of activity in the operators' breathing zone air during an 8 min interval following the loading of a washer, as indicated by two minute samples taken at 2 min intervals.

In the four cases tested, the activity fell off

Table 2. Operators' breathing zone air during various laundry cycles
(μc/ml)

Test group No.	Average loose counts/min per garment	Loading clothes into washer	Removing from washer placing in extractor	Removing from extractor placing in dryer	Removing from dryer	Pressing
1	575	2.4×10^{-10}	Background*	Background*	3.2×10^{-10}	Background†
2	710	3.6×10^{-10}	Background	Background	Background*	Background
3	3160	1.3×10^{-9}	Background	Background	Background	1.5×10^{-10}
4	8650	6.7×10^{-9}	7.8×10^{-11}	9.4×10^{-11}	Background	3.2×10^{-12}
5	16,300	4.8×10^{-8}	Background	2.7×10^{-10}	Background	1.4×10^{-11}
6	32,300	3.1×10^{-8}	1.3×10^{-9}	1.9×10^{-9}	3.5×10^{-9}	Background
7	77,000	9.0×10^{-8}	Background	Background	1.0×10^{-10}	Background
8	73,300	1.0×10^{-7}	1.3×10^{-11}	1.3×10^{-9}	1.0×10^{-10}	Background

* Background is 5×10^{-12} μc/ml.
† Background is 5×10^{-13} μc/ml.

Table 3. Airborne activity during washing and drying cycles
(μc/ml)

Test group No.	Center of wash room	Center of dryer room	Outdoors dryer discharge
1	7.2×10^{-12}	Background*	1.1×10^{-11}
2	Background*	Background	Background
3	5.4×10^{-12}	6.7×10^{-12}	2.1×10^{-11}
4	3.4×10^{-11}	2.3×10^{-12}	9.0×10^{-12}
5	1.0×10^{-10}	6.3×10^{-12}	5.3×10^{-11}
6	1.9×10^{-10}	7.1×10^{-11}	5.0×10^{-10}
7	3.0×10^{-10}	Background	1.0×10^{-10}
8	5.5×10^{-10}	1.0×10^{-12}	3.4×10^{-10}

* Background is 4×10^{-13} μc/ml.

Table 4. Decrease in activity of operators' breathing zone air
(μc/ml)

Test group No.	Contamination during loading of washer	Contamination 2–4 min after loading	Contamination 6–8 min after loading
9	1.0×10^{-7}	8.4×10^{-10}	2.3×10^{-10}
10	6.5×10^{-9}	1.7×10^{-10}	2.4×10^{-10}
11	1.0×10^{-7}	3.3×10^{-10}	4.6×10^{-10}
12	3.8×10^{-9}	3.9×10^{-10}	1.6×10^{-10}

very rapidly after the initial sample, indicating that this initial hazard is of rather short duration and of course, would be even less if more ventilation were available. It is obvious from these tests that the washer operator while loading the machine is exposed to significant quantities of airborne contamination at all times. During the loading of a washing machine with contaminated garments and for an additional short period following the airborne contamination ranged upward to a maximum of 10^{-7} μc/ml.

Using these data, and assuming one batch of wash per hour, integrated dust exposures were computed by time weighting the dust concentration at the breathing zone during the loading operation and adding to it the general room air concentration for the remaining part of the 60 min cycle (see Fig. 3). This plot shows a definite correlation between the amount of airborne exposure to the operator and the total activity in a batch.

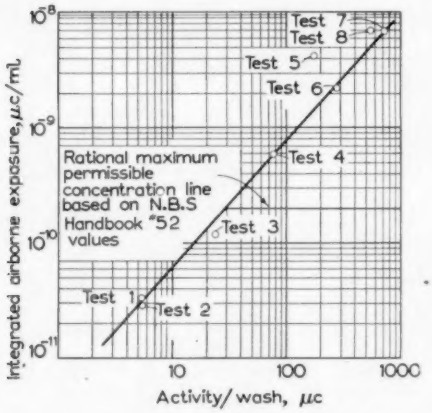


FIG. 3.

To locate a rational maximum permissible concentration line which might be suitable for a commercial laundry that could process clothing similar to that used at BNL, the following reasoning is suggested: National Bureau of Standards Handbook #52 states that the maximum permissible concentration of Sr^{90} in air for continuous occupational exposure is 2×10^{-10} μc/ml. For a 40 hr week this figure would be increased by a factor of 3. For non-occupational exposure this concentration must be divided by 10 making it 6×10^{-11} μc/ml. For a laundry operation where there is to be no detailed routine monitoring, the non-occupational limit would appear to be appropriate. Since most of the laundry at BNL contains not more than 20 to 30 per cent Sr^{90} this concentration might conservatively be increased to about 2×10^{-10} μc/ml. This figure would vary up or down of course depending on the composition of the contamination involved.

It is apparent from these tests that for this particular laundry there is an activity level below which contaminated garments can be processed without danger of exceeding maximum permissible airborne exposure. Test washes one, two and three could be considered safe.

Table 5 shows the activity of the wash water in μc/ml associated with washes of increasing activity. It can be seen that the maximum waste water concentration was 1.4×10^{-4} μc/ml.

Later work not reported here indicated that variations in wash cycle, detergents, etc., can affect the decontamination of garments to a considerable extent. For example, at the lowest levels of contamination tested, decontamination varied from a low of 39 per cent gross removal to a maximum of 65 per cent. This could in turn affect the amount of activity found in the wash water and the activity found in the effluent from the dryer.

Surface contamination

Long term operation has given the floor of this laundry a small fixed contamination which can not be removed without removing a surface layer of the floor. This level, while only a fraction of 1 mrep/hr, is sufficient to mask any other survey data which might reveal the contamination settling on the floor after a single operation. For this reason the smear paper technique was used to indicate contamination on floors and equipment. This was done by rubbing Whatman #41 filter paper across the surface and counting the contamination adhering to the surface of the paper. This method does not give an accurate measure of

Table 5. *Activity in the wash water*
(μc/ml)

Test group No.	Total est. μc/group using 2500 counts/min per μc	Concentration in effluent (μc/ml)	Total activity in effluent (μc)	Volume of water (gal)
1	14.7	3.2×10^{-6}	3.5	277
2	35	1.2×10^{-6}	1.7	375
3	43.3	6.7×10^{-6}	9.9	390
4	135	1.2×10^{-5}	13.0	287
5	224	3.7×10^{-5}	49.0	350
6	296	6.1×10^{-5}	89.0	386
7	738	1.4×10^{-4}	244.0	461
8	582	1.4×10^{-4}	219.0	413

the amount of activity, but it does give some indication of relative amounts of contamination falling on to the surfaces.

These smears did serve to show that while the extractor and dryer did not show contamination during these tests gross contamination was present on the washer and on the floor in front of the washer, with smears showing as much as 3×10^3 d/m.

Cross contamination

From two to four new garments never contaminated were added to all washes of contaminated garments. The amount of contamination remaining in the fabric of those garments after the wash is listed in Table 6.

Table 6. Counts/min *on new garments washed in machine with contaminated garments*

Test group No.	Av. gross counts/min per garment	Counts/min on clean garments washed
1	1360	91 and 79
2	3020	119 and 84
3	5280	218, 143, 133 and 262
4	10,850	1896 and 1944
5	19,300	1901 and 1804
6	35,300	1568 and 1940
7	80,000	4908 and 4020
8	76,300	6107 and 6517

For even the lowest levels washed, there was a residual contamination remaining in the clean garments. At the highest levels this residual amounted to several thousand counts per minute or as much as an estimated 2 μc per garment.

To determine the effect upon clean clothes washed in a machine which has previously washed contaminated garments, twenty-five new pairs of coveralls were washed immediately following the wash of test group #4. Of the twenty-five new garments, only five came out of the wash with a background count. Eleven showed up to 100 counts/min, three ranged from 100 to 200 counts/min, and six from 300 to 600 counts/min.

Throughout this work it was found that the reading on a geiger survey meter was almost entirely dependent upon the distribution of the contaminant upon the material, the reading being higher as the contaminant is concentrated. While spiking the garments, for calibration purposes, it was determined that the distribution had little effect upon the special monitor data. The garments in test group #4 showed the greatest discrepancy between the geiger survey instrument readings and the special counter used for these tests. Some of the garments in this group indicated almost no activity on the survey instrument while others showed readings as high as 7 mreps/hr.

Activity balance

If an attempt is made to balance the activity put into a wash with the activity remaining in the garments plus that found in the air and water samples, it can be seen that a large discrepancy exists. This remained somewhat a mystery until the exit air duct became partially clogged, causing the dryer efficiency to drop. The repair man took several cubic feet of badly contaminated lint from the duct. When this activity is added to that of the lint which did get through the dryer and blew into the outside air, it accounts for most of the apparently lost activity. It is clear that lint filters should be installed since readings of 20 mreps/hr or more were obtained with the geiger counter from small quantities of the lint.

CONCLUSIONS

The problem of whole body exposure received from laundering contaminated garments at BNL is a minor one even where garments of appreciable activity are laundered. However, the possibility of measurable exposure does exist, under some conditions, and film badges are considered appropriate.

The greatest hazard lies in the inhalation of airborne radioactive dusts. The data indicate that below a certain level one could send contaminated laundry to a commercial laundry without hazard to either the laundry operators or to the public. For an operation similar to that at BNL, garments containing as much as 1–2 μc each of mixed fission products could be handled safely without special precautions. Obviously conditions will vary widely with local facilities, techniques, isotopes, activity levels, work load, etc. Thus, the study presented here is only a start towards deciding what levels can safely be handled by a commercial laundry where no special precautions are to be taken.

It has been demonstrated that higher levels can be handled safely by careful management of the loading and sorting operations. The operator must understand the problem and must always be careful to make a minimum of dust. A laundry provided with good ventilation and special hooded facilities will also cut down tremendously on the dust hazard. If still higher levels are to be handled as with test washes 5, 6, 7 and 8, the use of a well fitting high filtering efficiency respirator during the period of handling will provide safe conditions.

In most nuclear laboratories and plants, respiratory protective devices, gloves, special protective suits, rubber shoe covers, etc., are provided for workers who may become contaminated. In general, the decontamination and servicing of this type of equipment can best be done at the lab or plant rather than by a commercial laundry. Since some washing facilities are required for this work it would seem advisable to pre-wash the heavily contaminated garments before sending them to a commercial laundry. In any event, the uncertainties are such that for appreciable amounts of activity an evaluation of equipment and procedure should probably be made before a commercial laundry gets into the radiation laundry business. However, it should be emphasized that if properly managed such work can be done safely.

The air vented to the outside of the building by the dryer does not constitute a serious hazard. However, significant activity is present (some samples showed above tolerance) so the re-use of this air for ventilation or heating purposes is not recommended. As mentioned above, the use of lint filters is advisable. This collected lint must be disposed of so arrangements should be worked out before operations begin.

The activity in the wash water showed no cause for alarm. The disposal of it requires some prior evaluation in terms of local conditions and the total amount of activity involved.

Contamination of the building and equipment will inevitably occur. If house-keeping and/or building construction are poor, the buildup of this contamination might well become a problem, since it is potentially a source of both whole body and airborne exposure and is in no way taken into account in this report. Surfaces such as monel or stainless steel, and smooth non-porous floors would help tremendously in keeping the area clean.

Standard geiger counter probe type survey instruments do not cover enough of the garment at one time to properly segregate the contaminated garments. At the present time no single instrument is known to be commercially available that adequately meets the requirements for this job.

Acknowledgments—Grateful appreciation is extended to E. SAWYER and E. WALSH of the Health Physics Division for their extra care and patience expended in the monitoring of the garments; and to E. BERGIN, W. MCCARTHY and A. WOODSON of Protective Clothing Service, Radiation Laundry for their cooperation in making the laundry facility available for these tests.

Health Physics Pergamon Press 1958. Vol. 1, pp. 184–188

SOME GENERAL CONSIDERATIONS IN CHEMICAL DECONTAMINATION

P. J. BARRY
Health Physics Branch, Atomic Energy of Canada Ltd.,
Chalk River, Ontario, Canada

(*Received* 31 *March* 1958)

Abstract—The elementary principles involved in the removal of radioactive contamination from surfaces are discussed. The properties of the materials used in washing formulae and their use in freeing cotton from different forms of contamination are described. The role of detergents and sequestering agents in preventing re-contamination until the wash liquor is removed is discussed in terms of elementary chemical theory.

THE object of any decontamination procedure is to remove loose activity and to reduce any radiation field associated with the contaminated surface to an acceptable level. The success of any decontamination procedure is measured by the extent to which these objectives are accomplished with a minimum change in other properties of the treated surface. Decontamination consists of two stages:

(1) The breakdown of the surface-contaminant complex.*

(2) The stabilization of the contaminant in the cleaning medium in the form of a medium-contaminant complex following its separation from the surface.

The extent to which either of these steps may be effected depends upon how firmly the contaminant is bound to the contaminated surface. Radioactive contaminants may be soluble or insoluble in water. Water-soluble forms may be ionic (electrically charged) or non-ionic (electrically neutral) while water insoluble forms may range in size from small molecular aggregates through particles of colloidal or submicroscopic to micro- and macroscopic dimensions. Both the electrical charge carried by, and the size of, the contaminating particle are important in determining the strength of the binding involved. Four binding forces will be considered:

(a) Electrostatic forces between charges of unlike sign.

(b) Physical forces other than electrostatic.

(c) Chemical bonds (i.e. where a definite chemical compound is formed between the reacting species).

(d) Mechanical entrapment.

(a) *Electrostatic forces*

All surfaces are electrically charged and the properties associated with the electrical charge at a surface assume more and more importance as the ratio of the surface area to the mass increases. Charged ions and particles are attracted to surfaces carrying opposite electrical charges and are held there by electrostatic forces. The molecules of the surface materials may themselves carry charged groups which contribute to the charge at the interface. For example, protein molecules carry either positive or negative groups depending on the acidity of the medium and thus can adsorb particles of either sign of charge.

(b) *Physical forces*

There are other binding forces between molecules and particles which give rise, for example, to the surface energy or tension at the surface of a liquid. The total energy of a

* Complex as used here refers to the union between the contaminant and the surface without specifying the nature of the union.

surface is proportional to its area and surfaces of materials which can flow assume a spherical shape which exposes the minimum surface area for a given mass. Surface energy is reduced if materials having a lower surface energy are substituted at the interface for adsorbed substances with a high surface energy. Small particles can coalesce and form large aggregates with a corresponding reduction in surface area.

The electrical charge on surfaces depends on the nature of the surface material and on the environment and can be altered very easily by changes in that environment. Thus, under suitable conditions, solid particles and water-insoluble liquids can be suspended in water and remain stable over long periods of time, e.g. cosmetic preparations. The suspension is stable as the particles carry like charges which repel each other and prevent contact. The suspension can be readily broken by adding suitable salts to the water phase. Following the adsorption of ions of sign opposite to the charge on the particle, the surface charge decreases and the particles no longer repel one another.

(c) *Chemical bonds*

A third binding mechanism in the contaminant–surface complex may be the formation of a chemical compound. Except at the surface, the component atoms of a metal are arranged in a repetitive pattern in such a way that all valency requirements are satisfied. At the surface however, this arrangement is no longer possible and the residual valencies may result in the formation of chemical compounds between atoms of the metal and those of its environment. An example is the formation of oxide films on metal surfaces. There may be chemically reactive groups in the molecules of such substances as proteins which will react with molecules in the environment to form chemical compounds on the surface.

In both these examples, the bond in the contaminant–surface complex has been formed to produce a stable arrangement with reduced free energy. To break the complex, an equivalent amount of energy has to be put back into the system. This energy may be electrical, chemical, thermal or mechanical and is often a combination of two or more of these forms.

(d) *Mechanical entrapment*

In most surfaces there are cavities of widely differing size and complexity in which small particles may be mechanically entrapped. The dislodgement of such particles can often be a formidable task and presents problems quite different from those discussed above.

THEORETICAL BASIS OF DECONTAMINATION

Let us first consider water-soluble contaminants bound to surfaces by electrical forces. Most surfaces in contact with a water solution of salts acquire a negative charge which attracts positive ions (or cations) from the solution. The important feature of this adsorption mechanism is that it is chemically non-specific, involving only the electrical properties of the two components. Thus, during decontamination a radioactive Sr^{90} ion may be replaced by a nonradioactive Ca ion or by two singly charged sodium or hydrogen ions. This type of exchange can be written in the form:

surface bound strontium + calcium^{++} (in solution) $\rightleftharpoons$ surface bound calcium + strontium^{++} (in solution)

The symbol "$\rightleftharpoons$" means that the reaction may proceed in either direction and the system will come to an equilibrium such that the amount of bound Sr^{++} will be determined by the relative concentrations of the two free ions. Thus, if the concentration of Ca^{++} in the solution is increased, a correspondingly increased liberation of Sr^{++} occurs from the surface. Alternatively, if the concentration of Sr^{++} in the solution is reduced, more Sr^{++} is liberated from the surface. This can be done by continually supplying fresh calcium solution so that Sr^{++} concentration in the washing liquor is always low. A more economical method is to add to the washing solution a sequestering agent. This is a chemical compound that reacts with the Sr^{++} and converts it to form which is not adsorbable. Many such compounds are known and have many other uses in chemistry. Some of these sequestering agents are sodium salts of organic acids containing several acidic groups. One such compound, "Versene" or EDTA, is the sodium salt of a compound carrying four

acidic groups. A sequestering agent of importance in laundering is "Calgon", the sodium salt of a polyphosphoric acid. In the presence of water this salt gives rise to two positively charged sodium ions and one negatively charged ion:

$$Na_6P_6O_{18}^{+} H_2O \rightarrow 2Na^{+} + Na_4P_6O_{18}^{=}$$

In the presence of Ca^{++} or Sr^{++} the phosphate anion incorporates the cation and so removes it from solution as a negatively charged ion:

$$Na_4P_6O_{18}^{=} + Ca^{++} \rightleftharpoons 2Na^{+} + Na_2CaP_6O_{18}^{=}$$

This reaction is widely used in softening water. The hardness of water is due to Mg and Ca ions which precipitate soaps as a scum. This is not only wasteful of soap but the coloured scum is re-adsorbed by the fabric, which then comes out bearing a yellowish cast. Of course other cations such as Sr^{++} and Ce^{+++} have a similar effect and if they are radioactive this process re-contaminates the fabric. Calgon prevents this from happening.

It is desirable that the compound formed between the cation and the sequestering agent should have a negative charge so that re-adsorption is inhibited. Re-adsorption would occur if either a positive or a neutral compound were formed. This may require that the surface be treated with oxidizing or reducing agents first so that the metal ion is converted to a higher or lower valency state in which form it will react with the sequestering agent to give an ion of the right properties.

Another property of sequestering agents is that they can often bring into solution insoluble compounds of the contaminant. Even insoluble salts in the water usually have a minute portion dissolved in the form of ions. If these ions are removed by sequestering, then more dissolve from the insoluble portion. This process continues until all of the insoluble substance has dissolved. Compounds of zirconium and niobium in very dilute solution precipitate to form colloids which are readily adsorbed on to surfaces. Oxalate ions sequester these metals in solution and so dissolve such adsorbed colloids.

Insoluble particles may also be "dissolved" by the use of detergents, such as soaps and the synthetic materials now available as soap substitutes. These detergents consist of two parts, a long chain water-insoluble portion to which is attached a strong polar or water-soluble group. One example is the soaps which consist of a long chain of 15 to 18 carbon atoms to the end of which is attached a carboxyl group in the form of the sodium or potassium salt. Such compounds are said to be surface active because they are adsorbed at the interface between two phases with a lowering of interfacial tension. This lowered surface tension results in the dispersal of insoluble particles. In addition, the charge carried by the surface active agent stabilizes the suspension by coating the surfaces of all the particles with charges of the same sign.

A process analogous to the dispersion of particles and their suspension in a liquid is the release by detergent of particles adsorbed on to a surface. The mechanism by which this is accomplished is not known, but stabilization of the separated phases is due to simultaneous adsorption of the charged surface-active molecules on to both particle and surface which, since they now carry like electrical charges, tend to repel one another. The washing of clothing in laundry practice is based on this property of surface-active agents.

The mechanism by which soaps and detergents perform their function is complicated and depends on the simultaneous operation of several factors. The importance of cleaning action of this type in decontamination processes is that very few surfaces in operational use are clean. They are coated with dirt and grease films and the dirt particles adhere to the surface very tenaciously. Such particles can readily fix radioactive contaminants. Decontamination then becomes a problem of freeing the surface of dirt and grease as in any other cleaning operation.

It is common experience that greasy surfaces are not wetted by water. The water runs into spherical globules and fails to come into intimate contact with the surface. In this situation, cleaning solutions are quite ineffective. Most surface active agents have the property of wetting a surface, some being more effective than others. The best wetting agents are not always the best detergents but it should be noted

that wetting properties are important in cleaning solutions to ensure that the cleaning agents may come into intimate contact with the surface. This is particularly important in the case of porous solids such as cotton where the actual surface is very much larger than the apparent surface area and the cleaning agents must penetrate into the fine pores to be effective.

The cleaning or decontamination process may be summarized as involving:

(a) The complete wetting of the surface and the contaminant by the solution.

(b) The breakdown of the surface–contaminant complex. This involves the expenditure of time and energy. It may require specific chemical pretreatment such as by acidic, alkaline, oxidizing or reducing agents to bring the contaminant into a chemical state where the surface complex is weakened.

(c) In general the reaction occurring during the decontamination may be represented as:

$$\text{contaminant–surface} + \text{solution} \rightleftharpoons \text{surface} + \text{solution–contaminant}$$

The more stable the contaminant–solution complex compared with the stability of the contaminant–surface complex the more efficient the decontamination and, what is equally important, the less will be the cross-contamination. The contaminant must therefore be stably held in the solution so that it may be freed from the surface. This is done by sequestering agents or surface-active agents, usually by both.

It will be possible to study these general principles in greater detail in the following section where we shall consider their applications in laundry practice.

DECONTAMINATION IN THE LAUNDRY

Most of the material handled by the laundry is cotton. Cotton is easily contaminated and it is not always so easily decontaminated. For this reason it has been suggested that other textiles or treated cotton fabrics that are not easily contaminated should be used. Nylon, for example, is resistant to contamination and is also readily washable. Water-repellent treatments of cotton likewise protect the fabric from contamination. However, the following considerations suggest that some of the seeming disadvantages of cotton may actually be advantages. Cotton has a high absorptive capacity for water compared with synthetic textiles. This not only makes it comfortable to wear but it also means that a spill of radioactive solution on a cotton protective coat is readily absorbed and confined within a relatively localized area. Nylon under similar circumstances would allow some of the spill to run off and contaminate other surfaces. The rest of the solution would soak through the garment with very little absorption of activity and would contaminate undergarments and the skin. Radioactive contaminants are readily adsorbed on cotton. They are held in a relatively small area while the water soaks into the fabric over a wider area and is soon evaporated. The activity is thus fixed and will not contaminate the air or other surfaces with which it may come in contact.

Cations held by ion exchange are easily removed by complexing agents. Contaminants held mechanically as insoluble particles precipitated in the fine pores of the fabric or adsorbed by dirt particles that firmly adhere to the fibres are much more difficult to remove.

We have evidence that the contamination which is difficult to remove from cotton is held by this latter mechanism. There are two things here which concern us in a general way. Firstly, cations held by exchange mechanisms may interact with washing-bath constituents and become converted from easily soluble forms to insoluble precipitates deposited well within the fine pores of the fabric. The polyphosphate sequestering agents present will help to prevent this. Secondly, the soaps and detergents used to remove dirt usually behave as dispersing or deflocculating agents, i.e. they break down the particles of dirt to finer size. The smaller the particles the greater the difficulty of keeping them from being adsorbed. Thus, while the detergent removes the dirt, it at the same time decreases the particle size and increases the chances of readsorption. Furthermore, the smaller the particle, the more tenaciously it adheres to fabrics so that it becomes very difficult to remove dirt that is smaller than about 0.1 μ. It is bad practice to allow clothing to soak for extended periods in detergent solutions

since this increases the amount of fine dirt particles redeposited on to the fabric.

There is a limit to the amount of dirt which may be removed from fabrics and so a limit to the amount of decontamination that can be achieved. We have discussed mainly contamination by small particles which are most likely to give trouble in the laundry. Other types of dirt such as grease, oil and waxes which may bind active material are usually fairly readily removed, but paint may require prior removal by a dry cleaning solvent.

Suitable laundry formulae can be devised from the principles discussed above. The removal of grease, oil and particulate dirt requires the presence of a soap or synthetic detergent. The removal of water-soluble ionic contaminants requires the presence of a cation-sequestering agent. Soap and detergents require alkaline conditions for their maximum efficiency. This rules out the use of acidic sequestering agents such as citric acid. The added alkali should have a high buffering power—that is, it should be able to neutralize the acidic components of the dirt without losing its alkalinity and, for this reason, sodium hydroxide is unsatisfactory. Alkalies such as sodium metasilicate ("Metso"), trisodium phosphate or modified soda (a mixture of sodium carbonate and bicarbonate) are extensively used for this purpose.

The polyphosphates such as sodium tripolyphosphate, Calgon, etc., are very suitable as metal-sequestering agents. These compounds contribute little to the alkalinity of the bath but they are compatible with it. They are often included in soap formulae as builders, i.e. they take no part directly in the detergent process but improve the efficiency of compounds that do. They are also used as water softeners.

Soaps in general are better washing agents than synthetic detergents provided the hardness of the water is not too great. The soaps combine two properties required in a cleaning agent, i.e. they have both good soil-removing and good soil-suspending properties. Carboxymethyl cellulose has been used to increase the suspending properties of a soap bath and it is therefore sometimes included in the formula as a builder.

After washing in one or more cleaning baths the clothing must be rinsed. Usually three or four soap or sudsing operations are employed. The first two function as cleaning baths, the remainder as rinses. This is necessary as the material absorbs large quantities of the wash solution that contains suspended dirt particles. Soaps must be added to the first rinses to maintain the conditions necessary for keeping the dirt suspended while the entrained wash liquor is progressively diluted. Subsequent rinses use water to remove the soap, alkalies and other salts. After several rinses the alkalinity of the fabric is considerably reduced but often another rinse in a weakly acidic bath is given to complete the neutralization. A final rinse is then given to remove any excess acid and the clothing is starched. Starching improves the appearance of the final product but in the decontamination laundry its main advantage is that it makes for easier removal of contamination picked up when the clothing is next used, as the starch itself is removed in the washing. Bleaching is commonly included in ordinary commercial laundry practice. It is not, however, to be recommended in the decontamination laundry as repeated bleaching degrades the fabric which increases the difficulty of later decontamination.

Health Physics Pergamon Press 1958. Vol. 1, pp. 189–191

A SHORT PROCEDURE FOR ELECTRODEPOSITING FISSIONABLE ELEMENTS

A. BRODSKY,* L. W. FAGG and T. D. HANSCOME
Nucleonics Division, Naval Research Laboratory, Washington, D.C.

(*Received* 12 *April* 1958)

Abstract—Electrodeposition seems to be the best method for producing thin, hard, adherent, and uniform deposits of fissionable elements.[1] Several procedures are available in the literature.[1,2,3,4] Some of these procedures were used by the authors in attempts to produce thin films of 100–400 μg U/cm^2 of platinum, but the yields obtained with these quantities were variable, and deposits were often flaky and uneven. Other procedures were too elaborate for the purpose at hand. The procedure given in this article was developed to afford an easy and reliable method of obtaining deposits of fissionable elements for use as neutron detectors.[5] It has been used to obtain thin, uniform deposits of uranium, thorium, and neptunium. All deposits were found to be hard and adherent enough to resist mild rubbing with forceps.

For the sixteen uranium deposits prepared by this method, the yields were measured radiometrically. The uranium deposited on the first disc with a yield of 74 per cent, and after a slight increase in the cell current, yields of from 87 to 100 per cent were obtained for the remaining fifteen deposits. Thus, modifications of this method may be useful for urinalysis of uranium.

After first coating an aluminium foil of about 5 in. × 5 in. with zincate solution as described by WILSON and LANGER,[3] the procedure given here was also used to cover the foil with a uniform layer of natural uranium having an α-count at 50 per cent geometry of 200 counts/min per cm^2. The deposit came down in high yield and was similar in appearance and quality to the uranium deposits on the platinum discs. This foil was made for use as an α-calibration source for measuring foils exposed in an electrostatic precipitator.

The details of the electrodeposition procedure are given below. The usual electrodeposition apparatus, as shown in Fig. 1, may be used in this procedure. For preparing larger foils, cylindrical anodes and cathodes may be used, with currents adjusted to give current densities of about 100 mA/cm^2 at the cathode.

PROCEDURE

Caution. It may be necessary that reagents be pure and free of metal ions, halides, or strong reducing agents. The authors found ACS grade reagents satisfactory. Containers, all parts of the deposition cell, and the platinum electrodes should be cleaned thoroughly with sulfuric acid–potassium dichromate cleaning solution and rinsed several times with distilled water.

(1) To prepare a deposit of 200 μg U_3O_8, for example, measure 0.2 ml of stock uranyl nitrate (2 g/l.) into a 30 ml beaker with cover glass. Add 2–3 ml of 4 : 1 $HClO_4$ to HNO_3 and place on hotplate in hood.

(2) Heat to temperature required to begin reflux of liquid from cover glass. Continue heating until volume decreases and dense white fumes begin to emerge. Control temperature to prevent too rapid evaporation, and remove beaker from hotplate while few tenths of a milliliter of liquid still remain. Set beaker aside to cool a few minutes. Then rinse bottom of cover glass with narrow jet of distilled water, allowing rinsings to fall into

* Present address: Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington, D.C.

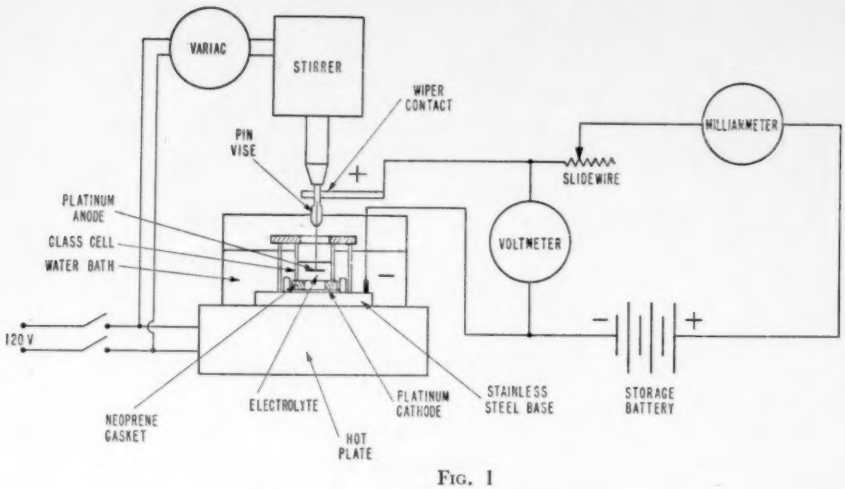


FIG. 1

beaker. Rinse sides of beaker, using rinse water sparingly to keep volume as low as possible.

(3) Add seven drops of bromcresol green indicator solution to beaker and neutralize carefully to blue endpoint with 1 M NH_4OH. Then back-titrate to green endpoint with 0.1 M HNO_3; pH should be from 5 to 6, but is not critical since buffer will be added.

(4) Pour contents of beaker (should be 20 ml or less) into deposition cell with subsequent washings. Add 5 ml of 0.4 M ammonium acetate to cell—this serves as a buffer to keep the solution from getting too acidic, increases conductivity, keeps the uranium-VI in solution, and helps reduce the solubility of the uranium-IV hydroxide at the cathode. According to COOK,[6] it is probably best to keep the molar ratio of uranium to acetate less than 0.05.

(5) Adjust the anode-stirrer so that disk is just below surface of solution. It is important that proper circulation be maintained and that most of the solution be contained between anode and cathode. Place cell in water bath and heat to temperature of 85°–90°C. Set stirrer speed at about 300 rev/min.

(6) Begin deposition maintaining water at 85°-90°C, and regulating voltage between 5–6 V and current density to about 100 mA/cm^2. With this current density, deposition should be continued for at least 4 hr if deposits of 200 μg of uranium are being prepared at good yield. Inspect occasionally during deposition; add ammonium acetate solution if needed to keep liquid level above anode.

(7) After deposition, add distilled water to cell, disassemble, and rinse deposit with a fine jet of distilled water. Dry under infrared lamp for about 20 min. Then for platinum disks, place disk in crucible and ignite to redness (about 800°C for Pt disk) for about 5 min. For aluminium foils pre-coated with zincate solution, an additional drying under an infrared lamp is sufficient.

Determination of yield

The amount of fissionable element deposited was determined from the equation:

$$W = \frac{C}{G \times A}$$

where

W = weight of fissionable element (μg)
C = counts/min of α-particles
G = geometry factor, corrected for backscatter and self-absorption; close to 0.5 in an internal gas flow type counting chamber
A = specific activity of element in terms of disintegrations/min per μg, using the following[7,8] values of A: 0.741 for U^{238}, 1.50 for natural uranium, 1519 for Np^{237}, and 0.247 for Th^{232}

REFERENCES

1. G. T. Seaborg, J. J. Katz and W. M. Manning, *The Transuranium Elements* Vol. II. McGraw-Hill, New York (1949).
2. A. C. Graves and D. K. Froman, *Miscellaneous Physical and Chemical Techniques of the Los Alamos Project.* McGraw-Hill, New York (1952).
3. C. R. Wilson and A. Langer, *Nucleonics* **11,** No. 8, 48 (1953).
4. C. L. Rulfs, A. K. De and P. J. Elving, *J. Electrochem. Soc.* **104,** 80–83 (1957).
5. A. Brodsky, L. W. Fagg and T. D. Hanscome, *Techniques for Using Fissionable Deposits in Neutron Measurements* NRL Report 4746 (1956).
6. A. O. Cook, *The Transuranium Elements* (Edited by G. T. Seaborg, J. J. Katz and W. M. Manning) Part I, Paper 3.5, p. 153. McGraw-Hill, New York (1949).
7. *Handbook of Nuclear Data* National Bureau of Standards.
8. G. T. Seaborg, J. J. Katz and W. M. Manning, *The Transuranium Elements* Vol. II, p. 1204. McGraw-Hill, New York (1949).

Health Physics Pergamon Press 1958. Vol. 1, pp. 192–204

NOTES

Converting a Gas Flow Alpha Proportional Counter to the Scintillation Type*

(*Received 6 December 1957*)

1. INTRODUCTION

WHEN it is desired to count nonconducting type samples in a gas flow counter, the effect of positive ion charge build-up on the sample can seriously affect the accuracy of the count. The geometry of the instrument will drop off at a rate dependent upon the activity of the sample. Also, with filter paper and smear paper samples, the loose material on the paper can be blown about during the flushing of the chamber, increasing the background and decontamination problems.

It was decided to convert a portion of the existing gas flow counters to scintillation counters in order to overcome the above objections and use these converted instruments exclusively for filter paper and smear samples. Since over twenty gas flow counters were to be converted, the most expedient and least expensive means of effecting the conversion was desired. A system of subassemblies and plug-in units was incorporated into the design so that once the several units were available in stock, the conversion could be completed in a few hours time and with a minimum of wiring changes in the original instrument.

2. GENERAL

The completed instrument is shown in Fig. 1. The "Manual Gas Purge" button and indicator light have been removed from the front panel. All other controls and the sample drawer are used in the same manner as in the original instrument. The location of the new plug-in pulse amplifier and the photomultiplier tube shield can be seen in Fig. 1.

The operating procedure for the scintillation counter is the same as for the original gas flow instrument with the exception that the gas purge cycle is eliminated. The sample can be counted immediately upon closing the sample drawer. An automatic high voltage shut-off is incorporated into the design to prevent extraneous light from damaging the tube when the sample drawer is opened.

The counting geometry varies with the source-to-phosphor distance, and can be adjusted over a range of from 25 to 50 per cent.

Assuming no contamination of the phosphor or surrounding metal surfaces, the background is primarily a function of thermionic emission noise in the multiplier phototube. If better than average 5819 tubes are selected, the background from this source can be kept under 10 to 15 counts/hr. The input sensitivity of the amplifier is variable over a range of from 0.4 to 0.05 V (see Section 6). This adjustment is located on the side of the amplifier chassis.

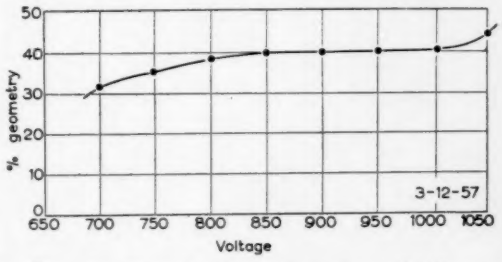


FIG. 2. Counting geometry vs. multiplier phototube voltage for the SC-1, SC-2 converted scintillation counter (source to phosphor distance $\simeq \frac{1}{16}$ in.).

3. PHOSPHOR PREPARATION

The phosphor is mounted on the Lucite light pipe of the photomultiplier tube. This same phosphor is used until it becomes contaminated. The phosphors are prepared by placing a 3 in. square of "Scotch" double-coated cellophane tape (No. 666) over a 2 in. dia. jar containing silver-activated zinc sulfide. The jar, with its Scotch tape "lid", is inverted and shaken

* U.S. Atomic Energy Commission Contract AT(29-1)-1106

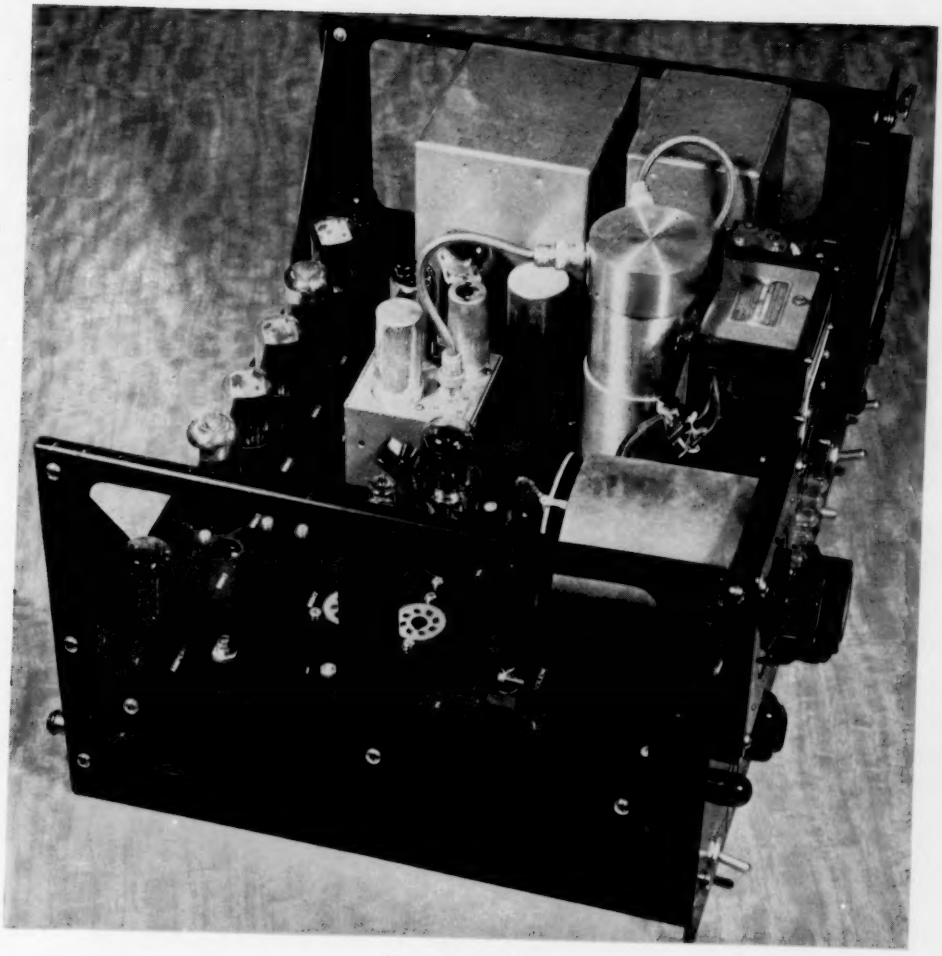


FIG. 1. Chassis.

for a few seconds. This deposits a very uniform layer of zinc sulfide on a 2 in. circle on the tape. The circle is cut out of the tape square, the protective coating removed from the remaining sticky surface, and fastened to the Lucite light pipe. In the event of phosphor contamination, the phosphor disc can be removed from the tube and a new one put in its place in a matter of minutes.

4. OPTICAL SYSTEM

The RCA 5819 photomultiplier tube is used here. To provide maximum optical efficiency, a $\frac{1}{4}$ in. thick Lucite light pipe is bonded to the face of the tube with DC-200 silicone fluid (1×10^6 centistokes). Care must be taken to eliminate air bubbles that may form in the silicone fluid. Any excess fluid that is squeezed out around the edge of the light pipe should be wiped away.

The photomultiplier tube assembly fits loosely in an aluminum shield. The chamber furnished with the original instrument is machined out and threaded to take the aluminum shield. The rubber "O" ring is retained and makes a good light seal between the planchet and the chamber.

The brass retaining ring is used to keep the phototube assembly from hitting the edge of the sample drawer as it is pulled out. When the phosphor is mounted on the Lucite, the ring is set so that the phosphor-to-sample distance will be that which gives the desired geometry when the piston is raised to the counting position. A blank disc of the proper thickness can be used to simplify this adjustment by placing it in the sample drawer, and pushing the ring down with the phototube until the Lucite comes into contact with the disc.

5. PHOTOMULTIPLIER TUBE CIRCUIT

The photomultiplier circuit is shown in Fig. 3. The voltage divider for the tube is designed after the RCA recommendations. With a supply potential of 900 V, the tube will be operating with 150 V between dynode No. 1 and cathode, and 75 V between each of the succeeding dynode stages. The current required from the high voltage supply by the voltage divider is approximately 15 μA.

The changes in the original high voltage circuit were found to be necessary due to the higher current required by the photomultiplier tube voltage divider. The use of one 10 MΩ resistor instead of three used in the original filtering circuit still provides very adequate filtering, and the voltage drop across this filter is much less troublesome.

A microswitch is placed at the rear of the sample slide, and connected in the primary of the high voltage circuit. This switch will open when the slide is pulled out, shutting off the high voltage so that in the event any light does reach the photocathode with the slide open, there will be no damage to the photomultiplier tube.

6. PULSE AMPLIFIER

The pulse amplifier of Fig. 3 has been redesigned from that of the original gas flow instrument. The plug-in feature was chosen to speed up maintenance in the event of amplifier failure. The operating temperature of the amplifier components is much lower in this design than in the original since most of the heat radiated from the amplifier tubes is directed away from the amplifier chassis components.

In the schematic, the 6J6 stage is a grounded grid amplifier, with a gain of about 9. The pulses from

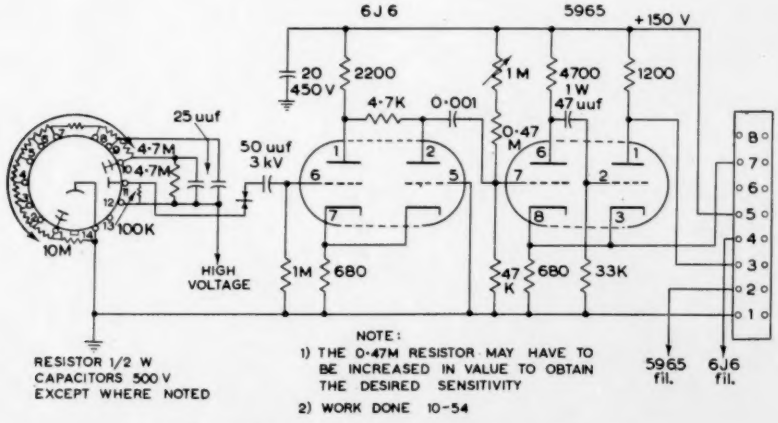


FIG. 3. SC-1, SC-2 Photomultiplier and amplifier circuit.

the photomultiplier tube range from 0.3 to 1.0 V in amplitude. This puts a pulse of from 1 to 9 V at the discriminator input. The 5965 discriminator is that of the original proportional counter with the exception that the computer-type tube is used.

The bias on the first grid of the discriminator is variable and is calibrated in terms of the input voltage pulse to the amplifier necessary to trigger the discriminator and produce an output pulse. The control is normally set to the lowest value of pulse that will trigger the discriminator before tube noise begins to increase the background counting.

7. CALIBRATION AND COUNTING

In counting with an alpha scintillation counter of this type, the effects of sample-to-phosphor distance and sample size must be considered. In using a source whose diameter is of the same order of magnitude as that of the phosphor, and with a finite distance between source and phosphor, the solid angle formed by any point on the source and the circumference of the phosphor disc is an important consideration. There will be different path lengths for the alpha particles to travel, hence different pulse heights appearing at the photomultiplier tube output. This means that the geometry of the instrument will be a function of source diameter as well as source-to-phosphor distance. It is essential, then, that the geometry calibration be made with a source whose diameter is equal to that of the sample to be counted. In so far as possible, the radioactive material on the calibrated source should have the same alpha energy spectrum as that of the sample to be counted. If the stainless steel spun top is used, the geometry can be adjusted over a range of from 40 to 50 per cent by moving the multiplier phototube assembly up or down inside its housing. By removing the spun top and placing the sample in the piston well, one can count irregular surfaced samples such as filter papers, smear papers and soil samples. The possibility of phosphor contamination from these loose material sample types is thus minimized. The geometry yield for this configuration will be from 25 to 35 per cent.

Health Physics Department, J. R. MANN
The Dow Chemical Company,
Rocky Flats Plant, Denver, Colorado

A Continuous Automatic Environmental Air Sampler

(*Received* 12 *November* 1957)

INTRODUCTION

THERE existed a need for an environmental air sampler which could aid in identifying and measuring the contamination of the environs around a reactor. The instrument in this report can be described as the type in which airborne radioactivity is collected on a fixed medium and then moved under a detector for counting and recording. The air sampler is automatic in that the air is filtered, a background is taken, and a decay of the previous sample is recorded on tape. Practically any time sequence is possible. The background may be eliminated if desired and the filter tape can be moved continually, or set to move automatically every 10 min or, with a small circuit change, once a week.

ANALYSIS OF DATA

A typical decay curve of the collected sample is plotted from the recording tape of the sampler as shown in Fig. 1. The instrument automatically runs a background after every complete cycle and the graph shows only the actual activity of the collected sample. The collected radioactivity from the environs for the most part is comprised of the solid daughter products of radon and thoron and the long-lived radioactive dusts. The detector has an added absorber of aluminum such that the alpha particles are not counted.

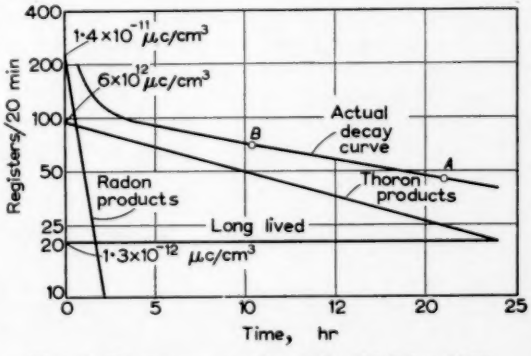


FIG. 1. Decay curve of sample, 25 May 1957, (Register = 16 counts) or (Registers − Bkgd. × 6.35 × 10^{-14} = μc/cm²) 9290−4103

Since the radon product has a short life it can be assumed that at points *A* and *B* (Fig. 1) the radioactivity is due mainly to the thoron products and the long-lived emitters. Points *A* and *B* are taken exactly 10.6 hr apart which is the half-life of the beta emitting thoron products. Therefore, it can be seen that the counts at point *A* are comprised of *X* counts from the thoron products and *Y* counts from the long-lived products while the counts at *B* are comprised of 2*X*

counts from the thoron products and Y counts from the long-lived products. By simple arithmetic we find that the counts from the long-lived products are equal to twice the counts at A minus the counts at B and the counts from the thoron products at point A are equal to the counts at B minus the counts at A. The thoron product and long-lived product decay curves are drawn and the remaining counts should then be due to the radon products. The subtraction is made and the remaining curve indicates a half-life

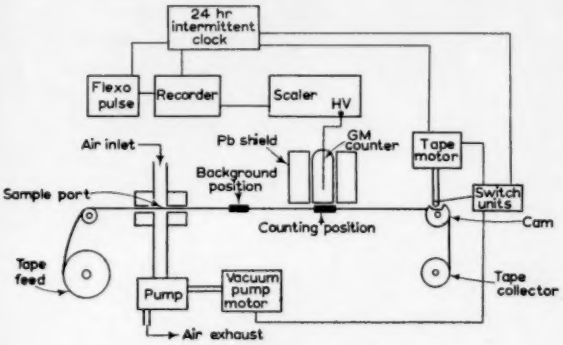


FIG. 2. Block diagram of an air sampler.

of about 28 min, which is approximately the half-life of the radon products. It can be seen then that analysis of the sample from this machine requires only two points on the recording tape to give sufficient environmental data. However, if short lived activity is suspected along with the radon and thoron products, the complete curve is available for study and the sample may be removed for further analysis.

Instrument description

The tape moving mechanism is comprised of a feed spool, driving rollers, a tape collector, and a 6 rev/min Bodine motor. When 24 hr samples are taken, one roll of filter tape is approximately 10 months' supply. The intermittent sampling is controlled by cams, a clock switch and a recorder relay. The cams are fastened to one of the drive rollers and determine the length of the period of the tape motor while the clock and recorder relay determine the beginning of the period.

A vaned rotary pump is used in which the vanes take up their own wear automatically for a uniform seal and pulseless air flow. An orifice reduces the air flow to 0.5 ft^3/min for 24 hr samples. By changing the orifice, up to 2 ft^3/min can be obtained for short periods.

The detection and recording system is comprised of a Nuclear Instrument and Chemical Corporation scaling unit, Model 162, a Streeter-Amet recorder, Flexo-pulse timer, and a detector which is approximately 30 per cent efficient for most beta energies. By the use of fast acting relays and low inductance coil the recorder will record up to 1000 impulses/min from the various scaling stages of the scaler.

Theory of operation

The operation can best be illustrated by Fig. 2. The collected sample moves to the detector in two steps. During the first step the old sample moves away from the detector and the new sample moves part of the way toward the detector. While in this position the instrument counts a background. After the background is counted the filter paper is moved to position 2 in which the collected sample is counted for the decay and a fresh air sample is collected. The counts can be taken and recorded every 10 min or up to every 2 hr. This also determines the length of time the background is taken. In Fig. 1, the sample was taken for 24 hr and the decay count was recorded every 20 min. The background count was also a 20 min count.

SUMMARY

An instrument has been described which determines automatically the amount of airborne radioactivity in the given volume of air. It has the following features:

(1) Individual samples can be obtained on various types of filter media from sampling periods of 30 min to 24 hr at a constant flow rate.

(2) Immediately after collection, the vacuum pump is shut off and the sample is moved into position in which either a background or a direct count is obtained and recorded.

(3) The counts from the radioactivity decay are fed into a scaler and register where the total accumulated counts which occur in any fixed time from 20 min to 2 hr are recorded.

Radiation Engineering, Atomics International
P.O. Box 309, Canoga Park, Calif.

R.L. KOONTZ
C.T. NELSON

Desert Basins and Waste Disposal

(*Received* 3 *December* 1957)

THE search for safety in waste disposal may follow either of two paths. One may seek to devise a method so certain that it can be used

anywhere, or one may seek a spot where there can be no hazard, no matter what method is used. In practice both method and environment must be considered jointly, but for a test site the inherent safety of the location must be paramount, for the method is under test.

THE epidemiologist uses the term *vector* for the agent which carries and transmits a disease. If we may borrow the term, there are two vectors which could carry and transmit radioactive wastes: air and water. Although some of the fission products are gases, the great bulk of the long lived isotopes which might get into our food chain are solids in solution in liquid wastes, and so the critical vector is water. Indeed, the critical vector may be limited to the ground water, for by the time appreciable amounts of activity have reached our lakes and streams we have already failed.

Granted the premise that we wish to experiment with large amounts of activity and that, in the event of the unexpected, transportation of contamination by ground water is the potential hazard, then our problem is to find a test site where this hazard may be disregarded. This may be done if it can be shown that: (1) all activity released to the ground will be permanently held by ion exchange, or that (2) the ground water is, without exception, moving so slowly that all the activity will decay while it is

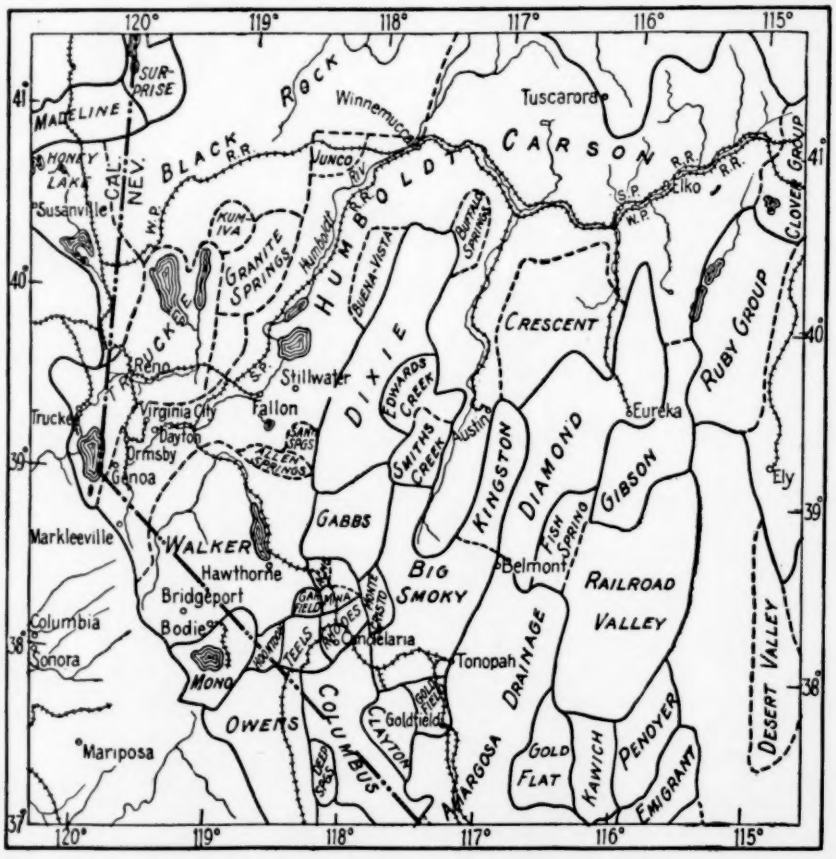


FIG. 1. Parts of Nevada and California, showing independent basins of internal drainage. From E. E. FREE. The topographic features of the Desert Basins of the United States with reference to the possibilities of occurrence of potash. *J. S. Dept. Agr. Bull.* 54 (1941).

still deep underground, or (3) if it can be shown that all the ground water is moving into, and will remain in, an area unoccupied by man or by any plants or animals which may enter into his food chain.

All of the long lived isotopes are adsorbed in varying degree by the soil, and in an area where the soil and rocks have a high exchange capacity they may be permanently trapped. But "may be" is not good enough, and reliance on ion exchange by the soil must be backed up by a long continued monitoring operation, which is expensive.

Calculations may suggest that the movement of the ground water in a particular area is so slow that activity may decay to the vanishing point before it reaches the world of life, but again this suggestion would have to be checked by long continued monitoring. No pretesting of the site could be as searching as the strain to which it would be subjected in the event of need; it is not possible to test a steam boiler with a bicycle pump.

Many of the desert basins in Nevada, and neighboring parts of adjacent states, are self-contained units hydrologically, for the little precipitation that reaches them is lost by evaporation (see Fig. 1). No surface or ground water flows out of them. This is not to say that each topographic basin is hydrologically isolated from its neighbors; some are interconnected underground, and on rare occasions surface water may flow from one to another where the divide is low.* The maximum limits of the possible movement of water cannot be taken directly from the surface forms, but it can be determined by test drilling and water level measurements, and in selected cases can probably be determined positively by a smaller investment of money and time, than would be required to clear any other type of site for large scale experimentation with high activity waste. Where this can be done, it should be possible to outline with certainty the total area in potential jeopardy, regardless of rates of movement, ion exchange or other factors which are difficult to measure. The only basic principal that must be relied on is that water always flows from areas of higher to areas of lower potential.

A typical desert basin may be 50 miles long and 16 miles wide (see Fig. 2). Most extend north and south, and are bordered on each side by mountain ranges that rise 3000–4000 ft above the desert floor. At the ends, the basins are closed by somewhat lower irregular ridges of hard rock, frequently faulted, and by alluvial deposits of clay, sand, and boulders, swept out by streams from the mountain ranges. If the basin leaks it is through one of these ends.

Most of the rain falls in the mountains, and on the higher peaks there may be snow in winter. This precipitation flows out on to the basin floor, commonly in the form of floods, for it is notorious that the drier the climate the more erratic the rainfall. Much or even all of the water soaks into the valley fill where the streams leave the mountains. Some may make its way over the surface to the lowest point in the basin, there to form a wide shallow lake, known as a playa. The ground water, fed by infiltration from the streams, flows slowly in the same direction, emerging as seeps or springs around the margin of the playa.

The playa is typically dry most of the year, or even for several years in a row if heavy storms are lacking. The water which does collect on occasion soon dries away, leaving the mud it carried as one more increment to the valley fill. Some of the springs or seeps near the margin of the playa keep small areas moist or even swampy the year around, but these do not flow rapidly enough to keep water standing on the lake bed.

The valley fill is composed of detritus swept in by the streams which drain the bordering mountains. It is typically coarse and poorly sorted near the mountains, and grades out irregularly into finer and better sorted material toward the playa. The nature of the fill is quite variable and at least in some of the basin the history is complex. The thickness of the fill in some of the basins is 1000 ft or more.

The ground water near the mountains along the margins of a basin is commonly fresh, and the water table commonly stands at a depth of 100 ft or more. As one moves toward the playa, the land surface slopes down more

* For an excellent general description of desert basins see N. M. Fenneman, *Physiography of Western United States*. McGraw-Hill, New York (1931).

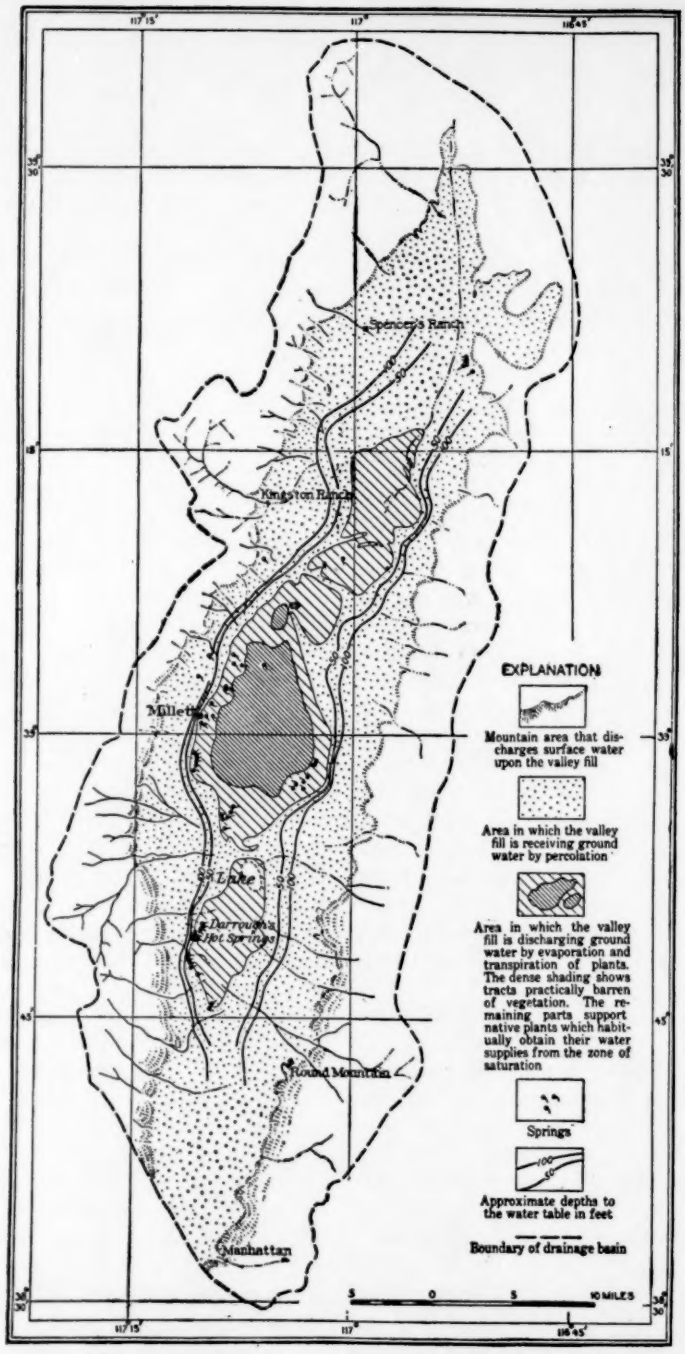

Spencer's Ranch
Kingston Ranch
Millett
Lake
Darrough's Hot Springs
Round Mountain
Manhattan
EXPLANATION
Mountain area that discharges surface water upon the valley fill
Area in which the valley fill is receiving ground water by percolation
Area in which the valley fill is discharging ground water by evaporation and transpiration of plants. The dense shading shows tracts practically barren of vegetation. The remaining parts support native plants which habitually obtain their water supplies from the zone of saturation
Springs
Approximate depths to the water table in feet
Boundary of drainage basin
5 0 5 10 MILES

FIG. 2. Map of northern drainage basin of Big Smoky Valley, Nev., showing the occurrence and circulation of water in the valley fill. From O. E. MEINZER, Geology and water resources of Big Smoky, Clayton and Alkali Springs Valley, Nev. *U.S. Geol. Survey Water-Supply Paper* 423 (1917).

rapidly than the water table, so that the depth to water is less. The water also is richer in dissolved salts. Deep drilling into the valley fill commonly finds water too highly mineralized to be drunk or used for irrigation, suggesting that circulation through the deeper valley deposits is very slow or nonexistent.

It would not be possible to work out in detail in advance the paths which would be followed by radioactivity reaching the ground water in such a basin. A study of water levels in the proposed test site basin, and in neighboring basins, could establish, however, that the movement of the ground water in certain parts of the basin had to be toward the playa lake, and could not lead out of the basin. The hazard, under these circumstances, would be limited to the basin. Any water used in one of these basins normally comes from the margins near the mountains, because only here is the chemical quality suitable, so that even the water supply, if there is one, is not necessarily in jeopardy. Many of the desert basins, of course, are now uninhabited and are largely or entirely government owned.

An unpleasant development, if it should materialize, would be the contamination of the clay on the playa floor, for this blows as dust during much of the year. But to reach the playa the contaminated ground water would have to move from the experimental disposal site to the springs or seeps, and careful site selection could make this a distance of some miles, as large a margin of safety as is now used to separate some of our major rivers from areas of active ground disposal. And we can afford to contaminate a playa or two if we have to. It is not proposed that the operation would be deliberately planned to do so, but rather that the precautions, monitoring and margin of safety could be substantially reduced as compared to those required in an area which drains into a region which is not expendable, or which produces directly or indirectly, a part of our food supply.

It is not my purpose to describe here the type of experiment which one might run in one of these basins, for my thesis is that such an area is inherently safe, no matter what is done. But as an illustration, it may be pointed out that an attempt to put radioactive waste down a deep well could be made with relative safety in such an environment. It would be possible to test the conditions under which such wells would plug, and also the methods for unplugging them. It would be possible to measure the temperature rise in and around such a well and the adsorption of activity by the aquifer matrix near the well. The operation could be at a comparatively shallow depth, which would appreciably decrease the time and expense required. This is not to suggest that the results obtained could be used as a criterion in regions of somewhat different geology, but much of general value will be learned from the first operation of this sort, just because it is the first.

There is always the possibility that the desert basin will prove to be a desirable place for actual disposal operations on a large scale, in which case one, perhaps the first, of our large fuel-element reprocessing plants might be located there. Clearly, if it lives up to expectation, the desert would be a safe place for such an operation. It would require somewhat longer shipment of the spent fuel elements, and construction costs would be a little high, but the possibility is worth considering. This is a second point, separate from, but related to, the question of selection of a test site.

We must also consider that there may be serious trouble locating a test site or a fuel element reprocessing plant, with its battery of storage tanks, in a more populated area, because of political objections. Politicians have a responsibility to protect their constituents, and some might object if fuel elements were shipped into their state for processing from power reactors in other states. This is not a problem now, but in the event of a well publicized accident, it could be serious. The Department of Agriculture had a difficult job finding a location for a hoof and mouth disease laboratory. But it is very hard to make a political issue out of rattlesnakes and sage brush.

It is not my contention that any desert basin could be used for waste disposal experimentation. Some carry main line railroads; some have mineral deposits of great value. The hydrology of many may be far more

complex than suggested by this brief discussion. But it may be possible to find one which has simple, basic, unique, and demonstrable advantages for such work, and so avoid the extensive pretesting and monitoring that would be required in areas of outward drainage.

I have been impressed by the objections of some of my hydrologist friends who have worked in arid areas, to the suggestion outlined above. A fundamental point, and one to which I wholeheartedly subscribe, is that danger lurks in the unknown, and that the geology and hydrology of the desert basins in Nevada are indeed poorly known. But we do not know the geology and hydrology of any area in the United States well enough to experiment there with the required degree of safety, and that required degree is far higher in most areas than it is in the desert basins. We may well know more about the ground-water hydrology of Long Island, and of the Brookhaven Laboratory area in particular, than of any other area in the country, and yet we could not use Brookhaven for a test site. We can probably reach the required degree of knowledge and safety in a desert basin, despite the complexity of detail, because the requirements are so much less.

A second objection has been that arid areas are those which need their water most. But any area must have a water supply, the need is only more apparent in the desert, not less real. But the value of water, and the emotional appeal of the life it brings are deeply ingrained in those who have made the desert blossom. As one of my friends said, "after spending a large part of my life trying to find water for irrigation in the southwest, how can I regard with favor a plan which might forever prevent the development of irrigation on one of these basins?" At least he realized that his reaction was dominantly emotional, for which I respect him. And it does not pay to under estimate the deep-seated nature of this reaction, for it has been part of our heritage. It was JONATHAN SWIFT who wrote in *Gulliver's Travels*; "And he gave it as his opinion, that whoever could make two ears of corn, or two blades of grass, to grow upon a spot of ground where only one grew before, would deserve better of mankind, and do more essential service to his country, than the whole race of politicians put together". And something of the same opinion can be found in the Bible, for many of those who wrote it were familiar with the desert. There are, indeed, many factors which render confusing the complex social as well as scientific problems which we must now try to answer.

But complex as these questions are, we cannot avoid them. In an early draft of an as yet unpublished paper, FLOYD CULLER, Director of the Chemical Technology Division of Oak Ridge National Laboratory, points out that the potential hazard in a fuel element reprocessing plant, and particularly in an ultimate waste disposal site, may be greater, ever far greater, than in any single reactor. For this reason he urges that overall hazard analyses be made of the decay cooling, radio chemical plant and waste disposal site system. He is concerned that such a study might show that the hazard is unjustified, for the cost of insuring against a single total catastrophe for a chemical plant or for a waste disposal area may be greater than the value of the power produced.

Because a carefully selected desert basin may provide a safe test area for determining this hazard with some accuracy, and may even provide a site which would materially reduce this hazard, it is our thesis that a study of these basins should be made.

Health Physics Division, WALLACE DE LAGUNA
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Maximum Permissible Radiation Exposures to Man[1]

(*Received* 16 *June* 1958)

INTRODUCTION

ON 8 January 1957 the National Committee on Radiation Protection and Measurements issued a Preliminary Statement setting forth its revised philosophy on Maximum Permissible Radiation Exposures to Man.[2] Since that time several of the NCRP sub-committees have been actively studying the necessary revisions of their respective Handbooks. These studies have shown the

need for (a) clarification of the earlier statement and (b) modification or extension of some of the concepts in that statement. Furthermore the International Commission on Radiological Protection (ICRP) has made minor changes in their recommendations.[3] Accordingly the NCRP has prepared a set of guides, given below, that will assure uniformity in the basic philosophy to be embodied in the several Handbooks. Since many of the Handbooks are followed closely in planning radiation operations in the U.S., and since the modification of a Handbook may require many months of effort, it seems wise to make the overall guiding principles available in advance of the reissuance of the revised Handbooks. These guides are not designed to take the place of any of the Handbooks; the principles given below will be extensively treated later in appropriate places. In the meantime Handbook revisions or supplementary statements will be issued as rapidly as possible.

Since the statement of an average per capita dose for the whole population does not directly influence the substance of the NCRP Handbooks, no further statements regarding such a number will be made at this time. In any discussion of the MPD it is impractical to take into consideration the dose from natural background and medical or dental procedures.

The changes in the accumulated MPD are not the result of positive evidence of damage due to use of the earlier permissible dose levels, but rather are based on the desire to bring the MPD into accord with the trends of scientific opinion; it is recognized that there are still many uncertainties in the available data and information. Consideration has also been given to the probability of a large future increase in radiation uses. In spite of the trends, it is believed that the risk involved in delaying the activation of these recommendations is very small if not negligible. Conditions in existing installations should be modified to meet the new recommendations as soon as practicable, and the new MPD limits should be used in the design and planning of future apparatus and installations. Because of the impact of these changes and the time required to modify existing equipment and installations, it is recommended on the basis of present knowledge that a conversion period of not more than five years from January 1957[2] be adopted, within which time all necessary modifications should be completed.

The basic rules and the operational guides outlined below are intended to be in general conformity with the philosophy expressed in the April 1956 statements of the ICRP.[3]

GUIDES FOR THE PREPARATION OF NCRP RECOMMENDATIONS

It is agreed that we should make clear distinction between basic MPD *rules* or requirements, and operational or administrative *guides* to be used according to the special requirements in any particular situation. Guides have the distinct value of retaining some reasonable degree of uniformity in the interpretation of the basic rules.

The risk to the individual is not precisely determinable but, however small, it is believed not to be zero. Even if the injury should prove to be proportional to the amount of radiation the individual receives, to the best of our present knowledge, the new permissible levels are thought not to constitute an unacceptable risk. Since the new rules are designed to limit the potential hazards to the individual and to the reproductive cells, it is therefore, necessary to control the radiation dose to the population as a whole, as well as to the individual. For this reason, maximum permissible doses are set for the small percentage of the whole population who may be occupationally exposed, in order that they will not be involved in risks greater than are normally accepted in industry. Also radiation workers represent a somewhat selected group in that individuals presumably of the greatest susceptibility (i.e. infants and children) are not included. However, for the persons located immediately outside of controlled areas but who may be exposed to radiation originating in controlled areas, the permissible level is adjusted downward from that in the controlled area because the number of such persons may not be negligible. With this downward adjustment, the risk to the *individual* is negligible so that small transient deviations from the prescribed levels are unimportant.

Controls of radiation exposure should be

adequate to provide reasonable assurance that recommended levels of maximum permissible dose shall not be exceeded. In addition, the NCRP re-emphasizes its long-standing philosophy that radiation exposures from whatever sources should be as low as practical.

DEFINITIONS

For the purposes of these guides, the following definitions are given:

Controlled area. A defined area in which the occupational exposure of personnel to radiation or to radioactive material is under the supervision of an individual in charge of radiation protection. (This implies that a controlled area is one that requires control of access, occupancy, and working conditions for radiation protection purposes.)

Workload. The output of a radiation machine or a radioactive source integrated over a suitable time and expressed in appropriate units.

Occupancy factor. The factor by which the workload should be multiplied to correct for the degree or type of occupancy of the area in question.

RBE dose. RBE stands for relative biological effectiveness. An RBE dose is the dose measured in rems. (This is discussed in the report of the International Commission on Radiological Units and Measurements, 1956. NBS Handbook 62, p. 7.)

I. BASIC RULES

(1) *Accumulated dose (radiation workers)*

(A) *External exposure to critical organs: whole body, head and trunk, active blood-forming organs or gonads:* The maximum permissible dose (MPD) to the most critical organs, accumulated at any age shall not exceed 5 rems multiplied by the number of years beyond age 18, and the dose in any 13 consecutive weeks shall not exceed 3 rems (I).*

Thus the accumulated MPD $= (N-18) \times 5$ rems where N is the age in years and is greater than 18.

Comment. This applies to radiation of sufficient penetrating power to affect a significant fraction of the critical tissue. (This will be enlarged upon in the revision of H59.)

(B) *External exposure to other organs: skin of whole body:* MPD $= 10(N - 18)$ rems, and the dose in any 13 consecutive weeks shall not exceed 6 rems (II).

Comment. This rule applies to radiation of low penetrating power. (See Fig. 2, H 59).

Lens of the eyes. The dose to the lens of the eyes shall be limited by the dose to the head and trunk ((A) above).

Hands and forearms, feet and ankles. MPD = 75 rems/year, and the dose in any 13 consecutive weeks shall not exceed 25 rems (III).

(C) *Internal exposures.* The permissible levels from internal emitters will be consistent as far as possible with the age-proration principles above. Control of the internal dose will be achieved by limiting the body burden of radioisotopes. This will generally be accomplished by control of the average concentration of radioactive materials in the air, water or food taken into the body. Since it would be impractical to set different MPC values for air, water and food for radiation workers as a function of age, the MPC values are selected in such a manner that they conform to the above-stated limits when applied to the most restrictive case, viz., they are set to be applicable to radiation workers of age eighteen. Thus, the values are conservative and are applicable to radiation workers of any age (assuming there is no occupational exposure to radiation permitted at age less than eighteen). The factors entering into the calculations will be dealt with in detail in the forthcoming revision of Handbook 52.

The maximum permissible average concentrations of radio-nuclides in air and water are determined from biological data whenever such data are available, or are calculated on the basis of an averaged annual dose of 15 rems for most individual organs of the body (IV), 30 rems when the critical organ is the thyroid or skin, and 5 rems when the gonads or the whole body is the critical organ. For bone seekers the maximum permissible limit is based on the distribution of the deposit, the RBE, and a comparison of the energy release in the bone with the energy release delivered by a maximum permissible body burden of 0.1 μg Ra^{226} plus daughters.

* Roman numerals in parentheses refer to notes at end of this statement.

(2) *Emergency dose (radiation workers)*

An accidental or emergency dose not exceeding 25 rems to the whole body or a major portion thereof, occurring only once in the lifetime of the person, need not be included in the determination of the radiation exposure status of that person (see p. 69, H59) (V).

(3) *Medical dose (radiation workers)*

Radiation exposures resulting from necessary medical and dental procedures need not be included in the determination of the radiation exposure status of the person concerned (V).

(4) *Dose to persons outside of controlled areas*

The radiation or radioactive material outside a controlled area, attributable to normal operations within the controlled area, shall be such that it is improbable that any individual will receive a dose of more than 0.5 rem in any one year from external radiation.

The maximum permissible average body burden of radionuclides in persons outside of the controlled area and attributable to the operations within the controlled area shall not exceed one-tenth of that for radiation workers (VI). This will normally entail control of the average concentrations in air or water at the point of intake, or rate of intake to the body in food stuffs, to levels not exceeding one-tenth of the maximum permissible concentrations allowed in air, water and food-stuffs for occupational exposure. The body burden and concentrations of radionuclides may be averaged over periods up to 1 year.

The maximum permissible dose and the maximum permissible concentrations of radionuclides as recommended above are primarily for the purpose of making the average dose to the whole population as low as reasonably possible, and not because of specific injury to the individual.

II. OPERATIONAL AND ADMINISTRATIVE GUIDES

(5) The maximum dose of 12 rems in any one year as governed by the 13 week limitation, should be allowed only when adequate past and current exposure records exist. The allowance of a dose of 12 rems in any one year should not be encouraged as a part of routine operations; it should be regarded as an allowable but unusual condition. The records of previous exposures must show that the addition of such a dose will not cause the individual to exceed his age-prorated allowance.

(6) The full 3 rems dose should not be allowed to be taken within a short time interval under routine or ordinary circumstances, (however, see paragraph 2 on emergency dose above). Desirably, it should be distributed in time as uniformly as possible and in any case the dose should not be greater than 3 rems in any 13 consecutive weeks. When the individual is not personally monitored and/or personal exposure records are not maintained, the exposure of 12 rems in a year should not be allowed; the yearly allowance under these circumstances should be 5 rems, provided area surveys indicate an adequate margin of safety.

(7) When any person accepts employment in radiation work, it shall be assumed that he has received his age-prorated dose up to that time unless (a) satisfactory records from prior radiation employment show the contrary, or (b) it can be satisfactorily demonstrated that he has not been employed in radiation work. This is not to imply that such an individual should be expected to routinely accept exposures at radiation levels approaching the yearly maximum of 12 rems up to the time he reaches his age-prorated limit. Application of these principles will serve to minimize abuse.

(8) The new MPD standards stated above are not intended to be applied retroactively to individuals exposed under previously accepted standards.

(9) It is implicit in the establishment of the basic protection rules that at present it is neither possible nor prudent to administer a suitably safe radiation protection plan on the basis of yearly monitoring only. It is also implicit that at the low permissible dose levels now being recommended, there is fairly wide latitude in the rate of delivery of this dose to an individual so long as the dose remains within the age-prorated limits specified above. In spite of a lack of clear evidence of harm due to irradiation at dose rates in excess of some specified level, it is

prudent to set some reasonable upper limit to the rate at which an occupational exposure may be delivered. Therefore, it has been agreed that the dose to a radiation worker should not exceed 3 rems in any 13 consecutive weeks.

(10) The latitude that may appropriately be applied in the operational and administrative control of occupational exposure will be dictated by two major factors (a) the type of risk involved and the likelihood of the occurrence of over-exposures and (b) the monitoring methods, equipment, and the dose recording procedures available to the radiation users. Where the hazards are minimal and not likely to change from day to day or where there are auxiliary controls to insure that the 13 week limitation will not be exceeded, the integration may be carried out over periods up to 3 months. Where the hazards are significant and where the exposure experience indicates unpredictability as to exposure levels, the doses should be determined more frequently such as weekly, daily, hourly or oftener as may be required to limit the exposure to permissible values.

(11) For the vast majority of installations (medical and industrial), operation is more or less routine and reasonably predictable and it may be expected that their monitoring procedures will be minimal. For such installations the protection design should be adequate to insure that over-exposures will not occur—otherwise frequent sampling tests should be specified. Where film badges are used for monitoring, it is preferable that they be worn for 4 weeks or longer, since otherwise the inaccuracy of the readings may unduly prejudice the radiation history of the individual. Where operations are not routine or are subject to unpredictable variations that may be hazardous, self-reading pocket dosimeters, pocket chambers or other such devices should also be worn and should be read daily or more often as circumstances dictate.

(12) Except for planning, convenience of calculation, design or administrative guides, the NCRP will discontinue the use of a weekly MPD or MPC (VII).

(13) The Committee has deliberately omitted the discussion of future exposure forfeiture for exposures exceeding the MPD on the grounds that any such statements might lend encouragement to the unnecessary use of forfeiture provisions.

III. NOTES

(I) The quarterly limitation of 3 rems in 13 weeks is basically the same as in H59 except that it is no longer related to the old weekly dose limit. The yearly limitation is 12 rems instead of the 15 rems as given in the NCRP preliminary statement of 8 January, 1957

(II) This is similar to the 1954 (H59) recommendations in that the permissible skin dose is double the whole body dose.

(III) This is basically the same as the 1954 (H59) recommendations on a yearly basis.

(IV) This is basically the same as the 1953 recommendations (H52).

(V) This is the same as the 1954 (H59) recommendations.

(VI) This is basically the same as the recommendations of 8 January, 1957.

(VII) This represents a minor change from the NCRP recommendations of 8 January, 1957, but no change in the basic MPD.

REFERENCES

1. This statement will accompany all copies of NBS Handbook 59, *Permissible Dose from External sources of Ionizing Radiation.* Copies of this statement may be obtained upon request from Publications Section, National Bureau of Standards, Washington 25, D.C.
2. *NBS Tech. News Bul.* **41,** 17 (1957); *Radiology* **68,** 260 (1957).
3. Report on the 1956 Amendments to the Recommendations of the ICRP. *Radiology* **70,** 261 (1958); *Acta Radiologica* **48,** 493 (1957); Brit. J. Radiol. **31,** 93 (1958).

National Bureau of Standards L. S. TAYLOR
Washington 25, D.C.

Health Physics Pergamon Press 1958. Vol. 1, pp. 205–206

LETTER TO THE EDITOR

Suggested Insignias for the Health Physics Society

(*Received 17 March 1958*)

THE attached insignia is submitted to you for your consideration with respect to potential use as a "Health Physics Insignia".

The significance of the design can be summarized in the following manner:

(1) The shield is symbolic of a protective mechanism against any aggressive agent or enemy (in this case ionizing radiation).

(2) The atom is clearly indicative of our general area of consideration.

(3) The nucleus, the heart of concern, consists of a problem and a solution.

(3a) The problem is the potential hazard associated with atomic energy products. This hazard is represented by the black horizontal portion of the cross. It is important that we recognize the problem and the hazard. Therein lies the significance of the black horizontal bar.

(3b) The solution to the problem is the safety provided and represented by the magenta* and yellow† vertical (upright) bar which symbolically eliminates the problem and the hazard.

(4) "Progressus cum salute"

The use of the Latin language was felt appropriate to express the international characteristic of the

* Black on figure.

† White on figure.

Health Physics Society. The words chosen were such that the English translations were relatively simple. The words "salute" or "salutis" are direct derivations from the word "salus" the ancient Roman goddess of safety. The words, "health and safety" may be directly interchanged in the Latin translation. Therein lies the significance of expressing the uniqueness of health and safety through the medium of a single derivative namely "salus." The choice of the preposition "cum" (with) was made to emphasize the equal relationship of progress and safety. The phrase, "progressus cum salute" (progress with safety) indicates in the positive sense the total economical worth of the health physics services to industry and the nation, and in the negative sense, it indicates that progress can not be made except with safety. A literary license was involved in the translation of the phrase Health Physics Society into latin. Strictly speaking the latin translation would be Societas Salutis et Physicorum. However, to simplify the translation, the English order of wording was applied and the conjunction (and) was omitted.

(5) Color

The shield is outlined in green and has a background of white. These colors are symbolic of the ability to proceed with safety of operations and indicative of the integrity and ethics of the health physics personnel providing the safety therein.

Considerable attention was given to the inclusion of the radiation warning insignia in the nucleus. This arrangement was, however, felt to detract from the specific use and meaning of the warning symbol. In addition, the placing of emphasis on the existence of hazards by the black horizontal portion of the cross is important and more directly applicable and of international significance to all laymen as well as health physicist. Therefore, I highly recommend that you consider the use of the *Health Physics Cross* in the final analysis of a Health Physics Insignia.

Acknowledgments—The author is deeply indebted to the following individuals for their contributions in the developement of the Health Physics Insignia above. W. H. TRURAN and Miss M. R. KENNEDY, KAPL; J. L. FLYNN, ALCO Products; T. J. BURNETT, ORNL.

General Electric Company J. J. FITZGERALD
Knolls Atomic Power Laboratory
Schenectady, New York

Health Physics Pergamon Press 1958. Vol. 1, pp. 207–211

NEWS

News Editor: SAUL J. HARRIS

Health and Safety in the Use of Radioisotopes

EXPERTS from ten countries have been convened by the International Atomic Energy Agency (IAEA) to assist in drawing up health and safety recommendations to govern the packaging, transport, use and disposal of radioisotopes.

The *ad hoc* expert panel started its work at the IAEA headquarters in Vienna under the chairmanship of IAEA consultants, Dr. GUNNAR RANDERS (Norway).

The IAEA is constitutionally required to establish and apply health and safety measures for operations carried out by it or under its auspices. Such provisions, after having proved their value in practice, would most likely also influence existing and new national legislation. The Agency already possesses extensive material on health and safety standards, either as applied or recommended, in many technically advanced countries. These data are now being analysed.

Health and safety codes will eventually have to be provided to cover a wide area, e.g. research and power reactors, chemical and metallurgical plants for processing nuclear fuels, plants for treating irradiated fuels, and mines.

It has been decided to give priority to work on radioisotopes as it is probably in this field that the Agency will first become operational; standards here are also likely to be somewhat less complex to establish than in the case of for instance reactors.

The task can be approached in three stages.

Scientific principles must be established. These will deal with the effects of various degrees of radiation on the health of man and permissible radiation levels. The Agency will draw heavily on the International Commission for Radiological Protection (ICRP), whose recommendations are generally accepted by world expert opinion as the soundest ones available at present.

After the scientific basis of radiation safety has been assessed it is necessary to translate the findings into practical requirements. These requirements must follow directly from the scientific principles. Advice and recommendations on how to meet the requirements would form a third stage.

It is proposed to work largely on the third phase at the initial meeting of the panel. As a result of the panel's work a manual of radiation protection practice should be available to assist the Agency's use of radioisotopes. The panel will also consider some of the more difficult problems connected with the drafting of a regulatory code.

IAEA's work on health and safety requirements will be co-ordinated with that of other international organizations, in particular WHO and ILO.

The experts invited by IAEA Director General, Mr. STERLING COLE, to participate in the work of the *ad hoc* panel are:

Dr. GUNNAR RANDERS (Norway), Chairman
Prof. BATCHKARIOV (U.S.S.R.)
Dr. H. J. DUNSTER (United Kingdom)
Dr. BERNHARD GROSS (Brazil)
Dr. F. HERCIK (Czechoslovakia)
Dr. H. JAMMET (France)
Dr. FORREST WESTERN (U.S.A.)
Dr. A. S. RAO (India)
Dr. ARNE HEDGRAM (Sweden)
Dr. FUMIO YAMASAKI (Japan)

Health Physics in the News

THE following items concerning Health Physics have appeared in newspapers, public press or technical journals during the last quarter.

Oak Ridge workers exposed

A criticality incident occurred at the Y-12 Plant in Oak Ridge on 16 June, 1958. Eight employees of the plant received significant exposures to radiation and were hospitalized for observation.

The accidental nuclear reaction occured at 2.05 p.m., on 16 June, in a 55 gal stainless steel drum in an enriched uranium recovery area at the Y-12 Plant. The recovery area is located in a wing of one of the large buildings at the plant.

On the basis of presently available information, the following was the sequence of events leading to the nuclear incident:

A portion of highly enriched uranium bearing solution leaked or otherwise flowed through a valved

pipeline from a small diameter tank in one wing of the building to another small diameter tank in the recovery area. The solution also filled piping which connects the first tank with two other small diameter tanks in the area. All of the tanks, because of their geometrical configuration, were of the "always safe" type and a nuclear reaction could not occur even though the tanks were filled with enriched uranium solution.

Subsequent to this inadvertent transfer of enriched uranium two of the three tanks in the recovery area were partially filled with water for routine leak testing following a monthly clean-out for inventory.

During this operation, an employee, unaware that enriched uranium had leaked or flowed into the tank and piping, drained the solution into a 55 gal drum. As the valve on the drain line was opened, the enriched uranium solution which was in the tank nearest the valve preceded the water into the drum.

The enriched uranium in the drum was then in a "non-safe" configuration and a nuclear chain reaction occurred. Although there was no explosion, there was an instantaneous emission of radiation.

Following the initial emission of radiation and upon water dilution, the material in the drum is believed to have pulsed in periodic emissions of radiation for a period not exceeding 4 min.

The initial emission of radiation activated the plant alarm system and plant emergency procedures were put into effect. Personnel evacuated the buildings and the main road to the plant was temporarily closed to traffic.

Radiation control specialists were brought in from other portions of the plant and other Commission plants in Oak Ridge to determine the source of the radiation.

By 5.00 p.m., radiation survey teams had established the drum as the source of the incident and defined the radiation field. The remaining portions of the building were then reoccupied.

Initial efforts following the incident were concerned with determining whether any individuals had been exposed to radiation, providing medical care, and to decontaminating and cleaning up the affected building wing.

Sheets of cadmium, a neutron-absorbing material, were inserted into the 55 gal drum to insure that further nuclear reactions would not occur, and samples of the material in the drum were taken for analysis. Later, the contents of the drum were transferred to "always safe" containers.

Eight employees were in the vicinity of the drum at the time of the incident. All of them were carrying out routine plant operations and maintenance. One employee, a chemical operator, was participating in the leak testing which inadvertently set off the reaction. He was within 3–6 ft of the drum when the incident occurred. The other seven were from 15 to 50 ft away at the time. Two were engaged in installing ductwork; two electricians were engaged in removing conduit; two others, a welder and a machinist, were located on a mezzanine near the scene of the incident; and the seventh, a chemical operator, was in the process of starting up an evaporator some 50 ft away.

Two of the men reported seeing a blue flash at the time the reaction occurred, and some of the men reported a strange sweet smell in the air. They all evacuated the area quickly upon hearing the radiation alarm.

Since the determination of the absorbed dose from a reaction of this type is complicated under any circumstances, special methods were used to determine the neutron and gamma exposure of the individuals nearby.

Indium foil which is incorporated in all Y-12 security badges was useful in identifying and verifying the exposed individuals. The absorbed dose was determined from the radioactivity of Na^{24} in the bodies of those exposed through blood and whole body counting and by measurements obtained through a mock-up of the accident. The latter series of experiments was conducted by the Oak Ridge National Laboratory.

Through use of these methods, it is estimated that the first collision absorbed dose received by these eight men was 365, 270, 339, 327, 236, 68, 68, and 23 rads respectively. The distribution of this absorbed dose as determined by the mock-up experiment was 77 per cent gamma, 23 per cent fast neutron, and less than 1 per cent thermal neutron.

The three men who received the lowest exposure were released from the hospital on 26 June. The five others who received the higher exposures remained at the Medical Division Hospital of the Oak Ridge Institute of Nuclear Studies where they underwent further observation before release on 30 July.

As of this date (about 9 weeks after the accident) there are deviations in the blood counts from that of the average individual and there is some loss of hair, but, otherwise, these five men are in satisfactory condition.

Contamination spill dramatized

In February 1958, *The Twentieth Century*, a nationally shown television program presented as its subject the reaction of employees, the public and the press to a spill of radioactive material in the M. W. Kellogg Company's Houston installation. A film print of the program entitled *Enter With Caution*, is

available through the Public Affairs Department, CBS Television, 545 Madison Avenue, New York.

International standards in progress

In June 1958, a meeting was held in Harrogate, England, of the International Organization for Standards' Technical Committee 85, Nuclear Energy. Subcommittee No. 3 of this committee deals with radiation protection standards. International agreement on a radiation hazard symbol based in principle upon the U.S. three-bladed propeller design was reached by another subcommittee of ISO/TC 85.

Sealed sources handbook issued

In June 1958, the National Bureau of Standards published NBS Handbook 66, *Safe Design and Use of Industrial Beta-Ray Sources*, approved January 1958 as American Standard Z-54.2-1958 by the American Standards Association. Other radiation handbooks issued by the Bureau during the reporting period include No. 64 *Design of Free-Air Ionization Chambers*, and No. 65 *Safe Handling of Bodies Containing Radioactive Isotopes*.

International isotope manual drafted

In June 1958, the International Atomic Energy Agency reported that its expert panel on health and safety in the use of radioisotopes had completed a first draft of a manual of safe practice for radioisotope users.

Emergency radiation services offered

In June 1958, the AEC announced it was "acquainting state and local officials with services the Commission and the Department of Defense have available should an incident involving radiation occur in their local area". The Federal Civil Defense Administration also reported that state and local civil defense groups will be instructed on "peacetime hazards from nuclear accidents".

Occupational Safety Conference highlights radiation

In March 1958, the President's Conference on Occupational Safety, conducted by the U.S. Department of Labor included a panel discussion workshop on *Radiation—A Controllable Hazard*, and also utilized "radiation purple" as the theme color for the conference.

Radiation and the states discussed

In May 1958, the National Association of Manufacturers and the National Industrial Council sponsored a two-day meeting on *Atomic Industrial Development—A Challenge to the States*, at which several papers were presented on radiation hazards. At the meeting, speakers called for "one set of basic exposure standards for the whole radiation field", commented that "so long as widespread distrust and fear of atomic energy is based upon the radiation hazard, management will face a solemn responsibility for the proper control of that hazard", and indicated that the recommendations of the National Committee on Radiation Protection "will ultimately be used as a basis for, if not physically incorporated into, most collective bargaining agreements in the field of nuclear energy".

State regulations summarized

As of the end of June 1958, 7 states (California, Connecticut, Massachusetts, Michigan, New York, Pennsylvania and Texas) had adopted comprehensive radiation protection codes and 14 states (Colorado, Connecticut, Delaware, Illinois, Kansas, Massachusetts, Michigan, New York, North Dakota, Pennsylvania, South Dakota, Texas and Wyoming and the new state of Alaska) had adopted registration requirements for hazardous radiation sources. A full summary of *State Activities in Atomic Energy* up through the end of June 1958 has been published by the Atomic Industrial Forum, 3 East 54th Street, New York 22.

Reactor fire safety codified

In May 1958, the National Fire Protection Association approved a *Tentative Recommended Fire Protection Practice for Nuclear Reactors* (NFPA No. 802T—1958). The guide provides information on nuclear reactors and indicates practices necessary for fire safety.

Federal radiation compensation urged

At an AFL-CIO Conference on Workmen's Compensation in April 1958, "special treatment at the Federal level" was advocated as "the only means of gaining necessary protection across the country" for employees in terms of adequate workmen's compensation coverage for radiation injury.

Reactor hazards placed in perspective

In May 1958, members of E. I. duPont's Reactor Engineering Section estimated that "the escape of 1.4 million pounds of unignited hydrogen sulfide over a period of half an hour is equivalent in hazard to a 1000-MW reactor that runs away and releases a cloud of fission products in which the decay heat is 2 MW.'.

Public Health Service radiation program centralized

In February 1958, the surgeon general of the U.S. Public Health Service announced the formation of an 11-member National Advisory Committee on Radiation. In April 1958, the Public Health Service announced it had appointed key officials to a newly created Division of Radiological Health under Dr. FRANCIS J. WEBER, as Chief. The Division is to be an organizational part of the PHS's Bureau of State Services.

Health Physics Society

Committees for 1958–59

Admissions

W. E. NOLAN (Livermore), Chairman
FLOYD HERR (Westinghouse)
H. A. MCCLEAREN (Savannah River)
R. J. MAGELL (Univ. of Calif.)
P. E. TOTH (Chrysler, Detroit)
E. J. EVANS (Canada)

Certification

E. E. ANDERSON (ORNL), Chairman
L. S. TAYLOR (NBS)
J. S. LAUGHLIN (Memorial Center, N.Y.)
W. A. MCADAMS (GE, Schenectady)
C. M. PATTERSON (Savannah River)

Exhibit

R. S. LANDAUER, JR. (Chicago), Chairman
G. C. CAIN (ORNL)
E. D. TROUT (GE, X-Ray, Milwaukee)
J. E. MCLAUGHLIN (NYO)

Finance

J. S. LAUGHLIN (Memorial Center, N.Y.), Chairman
R. G. GALLAGHAR (Liberty Mutual, Boston)
C. C. GAMERTSFELDER (GE, Cinn.)

Legal (Standby Basis)

C. R. WILLIAMS (Liberty Mutual, Boston), Chairman
S. HARRIS (New York)
W. A. MCADAMS (GE, Schenectady)
E. L. SAENGER (Ohio)

Local Arrangements

H. R. CRAFT (ORNL), Chairman
W. SNYDER (ORNL and Univ. of Tenn.)
A. SCHOEN (AEC, Oak Ridge)
G. R. PATTERSON (Y-12, Oak Ridge)
J. C. BAILEY (ORGDP, Oak Ridge)

Meeting Place (1960–61)

L. G. CHELIUS (Los Alamos), Chairman
J. F. EGE (Argonne)
J. C. GALLIMORE (Convair)
J. N. STANNARD (Univ. of Rochester)
C. A. ZIEGLER (Tracerlab., Boston)
G. C. BUTLER (Chalk River, Canada)

Membership

P. J. BARRY (Canada), Chairman
E. L. ALBENESIUS (Savannah River)
E. A. PUTZIER (Rocky Flats)
J. R. HORAN (Idaho Falls)
C. K. SPALDING (Boston)
W. A. HUNZINGER (Switzerland)

Nominating

J. F. SOMMERS (Idaho Falls), Chairman
R. H. BOUTELLE (Brookhaven)
H. P. YOCKEY (ORNL)
E. C. BARNES (Westinghouse)
G. WILCOX (Livermore)

Placement

C. E. HAYNES (ORNL), Chairman
R. M. BALTZO (Univ. of Wash.)
A. L. CARSTEN (Brookhaven)

1959 *Program*

G. V. BEARD (AEC, Wash., D.C.), Chairman
J. S. HANDLOSER (Brookhaven)
W. C. ROESCH (GE, Hanford)
E. NIELSEN (UCRL)
E. L. ALEXANDER (Industrial Reactor Labs.)
C. P. STRAUB (USPHS)
R. D. BIRKHOFF (ORNL)
E. J. STORY (N.C. State College)

Publicity

T. SAUNDERS (Kewaunee, Mich.), Chairman
J. R. HORAN (Idaho Falls)
W. K. KINGSLEY (Sandia)
C. CLELAND (Canada)
L. C. EMERSON (ORNL)

Standards

J. W. HEALY (GE, Hanford), Chairman
S. W. SMITH (NBS)
A. BRODSKY (AEC, Wash., D.C.)
E. D. GUPTON (ORNL)
R. G. MCALLISTER (Liberty Mutual, Boston)

Symposia (Standby Basis)

ALBERT L. SMITH (NRDL), Chairman
H. P. YOCKEY (ORNL)
T. S. CHAPMAN (Rocky Flats)
F. D. SOWBY (Canada)
P. M. BIRD (Canada)
C. G. STEWART (Canada)

Rules and Procedures

D. C. HUBBARD (AEC, Oak Ridge), Chairman 1959
J. C. HART (ORNL) 1961
F. L. PASCHAL (Convair) 1963
L. F. PHILLIPS (Brookhaven) 1962
W. C. REINIG (Savannah River) 1960

Graduate Program in the Health Aspects of Nuclear Technology

THE University of Pittsburgh has been awarded a $30,000 grant by the Atomic Energy Commission for the acquisition of equipment and instruments to be used in the teaching of nuclear technology as applied to the life sciences.

This Atomic Energy Commission award supports in part an expanded program of training in radiation health within the University's Department of Occupational Health. The program was recently established by a 10 year grant from the Rockefeller Foundation.

NIEL WALD, M.D. will head the Pittsburgh radiation health program faculty, according to ADOLPH G. KAMMER, M.D., Chairman of the Department. Dr. WALD, an internist, has had training and research experience in immunohematology and radiobiology. He served the Department of Radiobiology, USAF School of Aviation Medicine for two years. In 1954, he joined the National Research Council's Atomic Bomb Casualty Commission in Hiroshima as senior hematologist for three years.

Other members of this training group are HERMAN CEMBER, ALBERT A. SPRITZER, M.D. and JOSEPH A. WATSON.

This group is responsible for the radiation health programs on the campuses of the University and of the Carnegie Institute of Technology, as well as in certain industrial establishments and hospitals in the community. Members of the group are actively engaged in radiobiological programs supported by the Atomic Energy Commission.

Physicians and engineers who may be interested in advanced training in Radiation Health should address inquiries to the Secretary, Graduate School of Public Health, University of Pittsburgh, Pittsburgh 13, Pa.

Announcements

(1) The next meeting of the International Commission on Radiological Units and Measurements is being held in Geneva from 14 to 26 September, 1958. This will be a meeting of the members of the Main Commission and the Chairmen of the Committees. The individual Committees will not be meeting. It is anticipated that the report of this meeting will be available in the spring of 1959.

(2) The 1959 Nuclear Congress is scheduled to meet in the public auditorium in Cleveland, Ohio, 5–10 April 1959. The Health Physics Society is one of the sponsoring organizations and those health physicists wishing to present papers should submit summaries of 300–500 words to Mr. J. W. MCCASLIN, Secretary, Health Physics Society, Phillips Petroleum Company, Box 1259, Idaho Falls, Idaho. The deadline for these abstracts is 1 October 1958.

(3) An European Section of the Health Physics Society will be organized at Geneva, Switzerland, during the Second U.N. International Conference on the Peaceful Uses of Atomic Energy. The meeting will be held on the primaces of CERN. The tentative time of the meeting is the afternoon of 9 September. For detailed information, contact: W. G. MARLEY, Acting Chairman (GB) P. L. COURVOISIER, Acting Vice Chairman (Sw) W. A. HUNZINGER, Acting Secretary (Sw) B.M.Wheatley (CERN)

PAPERS TO BE PUBLISHED IN FUTURE ISSUES

L. S. TAYLOR: History of the International Commission on Radiological Units and Measurements (ICRU).

J. NEUFELD: On the relationship between the charge of an ion and its velocity.

J. M. DUNNING: Radiation exposures from nuclear tests at the Nevada test site.

R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD: The naturally occurring alpha-ray activity of foods.

B. R. FISH: Applications of an analog computer to analysis of distribution and excretion data.

C. W. MAYS, D. H. TAYSUM, W. FISHER and B. W. GLAD: Bremsstrahlung counting of Sr^{90} injected dogs.

I. C. NELSON: Electronic data processing and radiation exposure records.

FOREWORD

THE Editors have decided to print in this issue the abstracts of all papers submitted for presentation at the annual meeting of the Health Physics Society at Berkeley, California on 9–11 June 1958. Those papers not appearing on the programme are listed as read by title. The Editors feel that the publication of these abstracts will be a service to many members of the Society who may not have found it possible to attend the meetings, and will serve as a brief but permanent record of papers presented as well as an indication of areas of research. The Editors will welcome comment concerning the value of these abstracts to the readers of the Journal.

Health Physics Pergamon Press 1958. Vol. 1, pp. 215–234

THIRD ANNUAL MEETING OF THE HEALTH PHYSICS SOCIETY

9–11 June 1958 at Dwinelle Hall, University of California

Abstracts of papers presented at the meeting

Session 1

The industrial approach to radiological protection. L. F. TISCHLER, L. J. BEAUFAIT, JR., A. DE HAAN, JR. and GEORGE G. MANOV, Tracerlab, Inc., Reactor Monitoring Center, U.S.A.

The world-wide acceptance of the fruits of nuclear power and of radioisotope applications carries with it the responsibility by the user for the radiological protection of his facilities, his personnel and the surrounding community. The large growth in the number and in the variety of laboratories and industrial firms handling radioactive materials makes it increasingly important that practical measures be found to deal promptly with accidental, uncontrolled release of radioactive contaminants and that highly trained decontamination squads be organized and maintained in a state of readiness much as one now maintains a conventional fire fighting department. From the economic point of view, however, it is not always practical for each establishment to maintain its own squad and it is therefore necessary to rely on external assistance.

This paper discusses the formation and organization of an industrial mobile decontamination service from the points of view of the Federal and State regulations that must be met, the practical methods for decontaminating large structures, the protection of personnel during the operations, the safe disposal of radioactive wastes and the restoration of the premises to productivity. Typical cost figures and manpower requirements are presented.

Of particular importance is the close cooperation with the State and local officials and the preservation of satisfactory relations on the part of the community and the public toward the laboratory or plant in which the incident has occurred. The desirability of pre-operational site surveys to determine the normal background radiation levels with respect to future plant construction is stressed together with the advantages of incorporating preventive maintenance factors in the design and construction of the laboratory or plant. Illustrative examples are given.

The decontamination procedures described are based on factual experiences and are illustrated by photographs taken during clean-up operations exemplifying three typical accidents involving wide-spread contamination by radioactive materials: (a) long-lived gamma-ray emitters (cobalt-60, cesium-137, etc.), (b) neutron sources (radium–beryllium) and (c) alpha sources (radium).

Administration of nuclear safety at the New York operations office. JAMES E. MCLAUGHLIN, JR., United States Atomic Energy Commission, Health and Safety Laboratory, New York, New York.

Since 1954 the Health and Safety Laboratory has advised AEC contractors engaged in reactor fuel element fabrications on nuclear safety problems. At the present time, this service has been formalized into a standard review procedure to permit the Manager to concur in the safety of these contractor operations. Some typical fabrications are described in terms of the quantities and forms of special nuclear material processed and necessary nuclear safety procedures. Movement of material in a plant during concurrent fabrications, as well as storage, were of particular concern. Finally, an organization is suggested, which would permit adequate administration of nuclear safety without stifling production operations by excessively restrictive controls.

The stray-radiation field of the Bevatron.* ALAN R. SMITH, Radiation Laboratory, University of California, Berkeley, California.

Radiation survey work at the Berkeley Bevatron has been a continuous project of the Health Physics Group since start-up of the accelerator in November, 1954. A substantial body of survey data has accumulated, from which a general pattern for the stray-radiation field can be constructed. Included in this report is a summary of the characteristics of the radiation field pattern as currently understood, a description of the various techniques used to make radiation measurements, and a discussion of some serious problems encountered in survey work at this accelerator.

* Work done under the auspices of the U.S. Atomic Energy Commission.

A system for utilizing meteorological diffusion estimates. M. E. SMITH, I. A. SINGER, G. S. RAYNOR and R. BOUTELLE Brookhaven, National Laboratory, Upton, Long Island, New York.

Although considerable experimental evidence has shown the influence of meteorological parameters on the diffusion of particulates and gases, it has been difficult to utilize these complicated relationships in routine monitoring and

emergency procedures. Members of the Meteorology Group and Health Physics Division of Brookhaven National Laboratory have spent a number of years developing various components of a simplified diffusion prediction system. These developments have included instrumentation for the detection of radiation in the cooling air, and for the automatic conversion of wind records to quantities directly useful in diffusion. The mathematical basis for the system has come from analysis of diffusion data and the preparation of machine computation programs specifically for this problem. The system, as it has been applied to diffusion of the Brookhaven Reactor cooling air is described in detail, and the question of adaptability to other installations is discussed.

* Research carried out under the auspices of the United States Atomic Energy Commission.

External environmental radiation measurements in the United States. LEONARD R. SOLON, WAYNE M. LOWDER, ALBERT V. ZILA, HARRIS D. LEVINE, HANSON BLATZ and MERRIL EISENBUD, United States Atomic Energy Commission, Health and Safety Laboratory, New York, New York.

Recent interest in the dose to man from natural radioactivity has been stimulated by the assumption by geneticists of a linear relationship between radiation dose and the incidence of genetic mutation, although this evidence has not been demonstrated at the low dose rates prevailing in nature. However, the possibility of such a relationship motivates the suggestion that geographical variations in the frequency of spontaneous mutations may be correlated ultimately with differences in the radiation dose to populations.

The studies of the natural radioactive dose received by man can be divided into the dose from external and internal sources. Of the internal emitters, radium-226 and its daughter products are the most important of the natural nuclides, but the dose from these sources is delivered primarily to the skeleton and is of minor importance as far as the germplasm is concerned. The natural dose to the germplasm is primarily due to external radiation, although one internal source, potassium-40, does deliver a significant dose to the reproductive organs.

During the Summer of 1957, measurements were made of the external environmental radiations in numerous locations throughout the United States. Measurements were made with an ionization chamber inside an automobile under essentially identical field conditions of car loading and ionization chamber orientation, it having been established previously that the car attenuation did not affect the measured values in an important way (about 5 per cent).

The purpose of these measurements was to establish the approximate range of population exposures to the penetrating environmental radiation component, cosmic radiation and terrestrial gamma radiation, but excluding terrestrial beta radiation. The natural environmental radiation levels encountered ranged from a low of 8.4 microroentgens/hour along the Pennsylvania Turnpike to a high of 38.6 microroentgens/hour at the summit of Pikes Peak.

Among the major United States cities visited, Denver, Colorado exhibited the highest radiation levels with a range of 16.6 to 22.4 microroentgens/hour with an average of 18.5 ± 1.5 microroentgens/hour.

These measurements were made during part of the period of Operation Plumbob, the Summer 1957 series of United States continental weapons tests at the National Test Station in Nevada and these tests influenced the measured dosimetric values to some extent. Elevated levels were encountered in eastern Arkansas and the Black Hills of South Dakota. That the initial elevated levels were attributable to fallout was demonstrated by the reduction in the measured levels by factors of two to four upon resurvey.

A resurvey of the Denver area almost three months later furnished results essentially identical with the earlier survey indicating that recent weapons test debris had no measurable effect on the environmental radiation level in Denver during the initial survey.

Measurements were made with a 20-liter ionization chamber filled with air at atmospheric pressure. The resulting ionization current was measured with a vibrating reed electrometer connected as a continuously reading voltmeter driving a pen recorder. To suppress completely the beta response, the chamber was positioned in an aluminum frame so that including the polyethylene wall, the gas volume was enclosed by 1.08 g/cm^2 of material corresponding to a Feather range of a 2.26 MeV beta particle.

As is well known, minute alpha contamination in an ion chamber at atmospheric pressure can produce an ion current which is of the same order as the ion current being measured. For this reason it is important that the effect of the contamination be measured or that the alpha produced current be suppressed. Several different methods have been used by previous investigators.

In our measurements we have resorted to a technique which relies on the difference in electric fields necessary to effect total collection of ion pairs produced by particles of low and high specific ionization, i.e. electrons from gamma or cosmic ray interactions and alpha particles respectively.

The results of this environmental radiation survey is compared with the estimates of other investigators; including the measurements of Sievert and Hultqvist; Hess; Neher; Compton; and the calculations of Burch and Libby.

Environmental hazards associated with the milling of uranium ore—a summary report. W. B. HARRIS, A. J. BRESLIN, HAROLD GLAUBERMAN, and M. S. WEINSTEIN, Health and Safety Laboratory, U.S. Atomic Energy Commission.

The financial incentives for rapid build-up of a large stockpile of uranium materials in the United States, Canada and Britain has resulted in an enormous increase in the number of individuals directly associated with their processing. Because of the relatively low grade of ore

which is normally encountered in nature, the production of this large volume of material requires mining and processing of approximately 20 million tons of ore a year. The magnitude of such an operation is obvious, as is the number of people involved. Inasmuch as several countries are in the competitive market for the supplying of these materials, there is great stress on the dollar economy of production. This results in a production oriented situation which is vastly different from that which exists in the laboratories and production plants which are supported by AEC cost type contracts. Consequently, many employes are exposed to concentrations of radioactive materials which would not be tolerated by AEC contractors.

In order that a sound engineering base be evolved for providing adequate health protection to the large number of workers in this field and to assist the AEC procurement and inspection organizations in evaluating expenditures for engineering controls of health conditions in the milling operations, a survey was performed which covered the 12 domestic mills which were in operation as of July 1, 1957. This report presents a detailed analysis of the daily average exposures of all plant personnel to the materials of greatest interest in this kind of processing. Other items which were covered in the survey include particle size studies, dust and direct radiation measurements outside of the plant and radioactive materials in effluent and ground water streams.

Several recommendations are included. These cover a suggestion for a simplification of the interpretation of the maximum permissible concentration in ore handling areas, modifications in some of the procedural practices and a general format for ventilation requirements for various specific process areas.

Session 2A

Radiological protection considerations for a plutonium fuel element fabrication pilot plant.* A. J. Stevens, Washington.

A successful demonstration of a plutonium recycle test reactor requires the investigation and demonstration of several types of fuel elements both to gain information and to determine the economic optima. Certain characteristics of these fuel elements necessitate investigation such as geometry, metallurgy and jacketing techniques. Although exact specifications for the advanced fuel elements are not defined, the equipment in the plutonium fabrication pilot plant performs typical operations such as rolling, swaging, drawing, forging, welding and other metallurgical operations on prototype metallic fuel elements. In the case of ceramic fuel element prototypes, other operations prevail such as pressing, sintering, sizing and grinding. The building is designed for contamination control as well as containment and provides safeguards for the protection of personnel. To control contamination spread, in the event of an incident, the building is divided into two zones. Zone I the "cold" zone constitutes the offices, lunchroom, change rooms, restrooms and cold laboratories; this zone contains no plutonium or other radioactive materials. Zone II is the "hot" side or process area. Zone II is used for jacketed or unjacketed plutonium, its alloys and compounds. Consequently, zone II has a considerable contamination potential. The building heating and ventilation system is designed to minimize air-borne spread of contamination within the structure, particularly from the "hot" to the "cold" zones. Because plutonium is an extremely toxic material, personnel working with or in the vicinity of the material must be adequately safeguarded against its harmful effects. This element is also reactive and many of its alloys are extremely pyrophoric, even at room temperatures. This requires that any work on the unjacketed metal or alloys be performed within a hood to control plutonium contamination and in a high vacuum or inert gas atmosphere to control oxidation and other reactions. An air sampling system is provided for the pilot plants to allow radiological monitoring of the building atmosphere. Air sampling monitoring stations are provided in all of the work locations of the controlled area where high plutonium hazards could prevail or incidents might occur. Additional air sampling stations are provided at strategic points in the controlled area exhaust systems including stations at the final exhaust plenum to the stack. With this provision, it is possible when an air sample result is above the MPL at the stack to trace back through the duct system and determine which hood or operation's filter is at fault. Central vacuum producing equipment is installed with adequate controls to insure uniform airflow rates at the air sample monitoring stations. The air sampling vacuum manifold is calibrated. Air sampling of other areas is accomplished by portable air sampling equipment. Radiation monitoring in the building consists primarily of alpha particle detection. Portable neutron and gamma monitors are provided to facilitate the monitoring of operations on neutron and gamma emitting materials. Fixed alpha detection monitors, with probes, are located at the exit of each potentially contaminated work area for personnel surveys. Hand counters are located near the building exit. Portable alpha detectors are also assigned to special locations for particular applications.

* Work performed under Contract No. W-31-109-Eng-52 between the Atomic Energy Commission and General Electric Company.

Preliminary efforts towards the establishment of surface contamination standards. L. C. Emerson, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

The paper summarizes the work of an American Standards Association Committee towards the development of criteria for establishing permissible levels of contamination for industrial materials. This is a controversial subject and one which, because of a serious lack of uniformity, has lead to administrative difficulty with both operational and managerial groups.

One of the principal efforts of the committee has been directed towards the establishment of a base upon which surface contamination standards can be logically developed. The paper discusses the results of experimental data relating surface contamination to human exposure potential. "Fixed" contamination is discussed from the standpoint of the associated external hazard and "loose"

contamination from the internal hazard standpoint. The methods by which contamination limits can be derived and the results of calculations based on available experimental data are presented.

Some discussion is included relative to both prompt fission products released in nuclear explosions and radionuclides produced continuously by reactors. The special problem of labeling requirements for products containing radioactive materials is also discussed.

Fission break monitor for gas cooled reactors and experimental systems.* J. F. SOMERS and I. J. WELLS, Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.

This paper presents a sensitive method of monitoring the effluent stream of a gas cooled reactor or experiment for the presence of fresh fission products in the effluent stream. The system is simple, inexpensive and can be calibrated to indicate the rate of evolution of fission gas molecules from a fuel element if the rate of evolution is constant or changes slowly in relation to the short half life fission product gases.

* Work done under contract to the U.S. Atomic Energy Commission.

The application of time and motion studies to exposure conservation. GEORGE LEWIS HELGESON, General Electric Co., San Jose, Calif.

In routine or semi-routine operations involving exposure to radioactive materials, substantial savings in personnel radiation exposure may be realized by the application of time and motion study principles to exposure problems. These principles are especially useful for determining those jobs or job steps which contribute the major portion of the exposure to the overall operation. An electronic exposure integrator is described which has proved to be a valuable tool in determining the exposure per job step. The distribution of exposures for a particular job or individual does not follow a Normal, or Gaussian distribution but may be made to fit the Normal distribution by a logarithmic transformation. One of the primary conclusions one draws from these types of studies is that through proper education and on-the-job training each man may reduce his own exposure to a new low value through the use of improved handling techniques, remote handling mechanisms, jigs for holding radioactive sources, shields, etc. Another benefit is that production can be increased where exposure formerly was a limiting factor.

Iodine monitoring at the national reactor testing station. C. W. SILL and J. KENNETH FLYGARE, JR., Analysis Branch, Health and Safety Division, Idaho Operations Office, U.S.A.E.C.

Investigation of new types of experimental fuel elements may require that they be operated beyond the point of failure to determine maximum operating characteristics. The fission products released must be monitored positively and immediately for the protection of personnel and property in surrounding areas. Unless the test element was operated for a considerable time before failure, the isotopes of iodine will be of prime importance from the standpoint of radiological dose. There has been considerable argument as to the economic and practical feasibility of using a filter for large air-cooled reactors partly because of uncertainity as to the physical state of iodine as released from a ruptured fuel element during operation. The same question must be answered before adequate sampling measures can be devised for analytical control. Accordingly, samples were taken directly from the stack after such a rupture had taken place. The air stream was drawn through two molecular membrane filters and two alkaline thiosulfate scrubbers connected in series. Analysis showed that the iodine activity was predominantly in the first scrubber when the dust loading was not excessive. In some tests in which the use of chemical fuels contributed a high concentration of carbon particles in the stack, more than half of the total iodine activity was found on the first filter. It is concluded that the iodine released from an operating fuel element is predominantly gaseous, and its collection on a filter will depend on its having been adsorbed on a dust particle during its previous history. It is expected that iodine will generally not be collected on mechanical filters. The recent experience at Windscale in England supports this expectation.

When a spent fuel element is dissolved in nitric acid at the beginning of chemical processing for the recovery of unfissioned fuel, fission products that are gaseous under the conditions employed will be evolved in the off-gas stream. If the fuel elements were short-cooled at the time of dissolution, large quantities essentially pure isotopes of iodine will be released. Evacuation of either on-site or off-site personnel for the period of operation is expensive. Waiting until after the fact to evaluate the actual thyroid dose received by personnel is intolerable. Liquid scrubber solutions require too much time before laboratory analyses can be made and answers received in the field and filter paper samples are inadequate because of the gaseous nature of the iodine. A continuous monitor has been developed that permits efficient collection and immediate detection of iodine activity, keeps track of the total accumulation of iodine with time and presents the data as a permanent written record. Iodine activities that will result in a total integrated dose of 5 mr to the thyroid can be detected easily. Personnel may then be allowed to remain where they are until the thyroid dose reaches a preassigned level before evacuation need be undertaken.

The monitor is composed of a plastic cylinder filled with activated charcoal through which air is drawn. The cylinder is placed in a lead-shielded, thallium-activated sodium iodide well crystal connected to either a linear or logarithmic counting-rate meter. The output of the latter is fed to a recording instrument which plots activity as a function of time. A field cartridge is also in use for monitoring outlying field locations where time is not so important and where the large number of sampling stations required will not permit the expensive auxiliary equipment used with the continuous monitor to be duplicated at every station. The field cartridge also incorporates

a filter so that particulate and gaseous activities can be separated if desired. Both the field cartridge and the continuous monitor can be used for monitoring other gamma-emitting species.

Some unusual radiation dosimetry problems associated with radar installations. R. ZENDLE, G. E., Schenectady.

High powered radar presents a rather unusual health physics problem; to measure X-radiation dose-rates from high frequency transmitters with conventional survey instruments normally sensitive to rf interference.

The rf interference can usually be prevented by enclosing the instrument with wire screening; however, this introduces errors as high as 60 percent due to absorption of soft X-radiation in the screen. Correction factors have been found experimentally for X-ray energies up to 140 KVP for 80 mesh copper screening.

There is also the possibility of errors caused by the pulsed nature of the X-ray source because of space charge and recombination effects in the ion chamber. It is shown that a conventional Cutie Pie survey instrument would read without significant error up to about 5 rad per hour provided that the duty cycle is greater than 0.05. On the other hand, with duty cycles in the order of 0.001 (1 μsec pulses at a rate of 1000 pulses per second) higher chamber voltages are required to maintain linearity over the full range of most instruments.

Angular distribution of weapons radiation: application to radiation shielding. R. H. RITCHIE and G. S. HURST, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

The Health Physics profession is concerned with the protection of man from the deleterious effects of ionizing radiation. Limits of exposure to these radiations can best be determined by the study of humans under conditions which permit accurate measurements of dose and effects. The group of exposed individuals in Hiroshima and Nagasaki offer the best opportunity for this purpose. A long-term study of medical effects is in progress at the NRC-AEC operated Atomic Bomb Casualty Commission in Japan. A critical study of the dose received by these individuals has just begun in this country.

The dosimetry study, which is a joint Oak Ridge National Laboratory–Los Alamos Scientific Laboratory-Air Force project, has become known as the "Ichiban" program. It is concerned with an accurate determination of the doses received by those individuals who were exposed in light residential structures in the two cities. The first phase of the program consisted of participation in the 1957 continental weapons test series.

There were three main divisions in this work. First, basic measurements of the angular distribution of radiation neutrons and gamma rays in open air at various distances from selected nuclear devices were made. These measurements were accomplished by means of large water- and lead-filled collimators which allow radiation from a limited angular interval to strike the dosimeters used. Second, exploratory attenuation measurements were made on typical structural materials. Third, the total attenuation of two light frame houses was determined for the radiations from one nuclear device.

The results of these measurements and their implications in the determination of the doses to those exposed in Hiroshima and Nagasaki will be discussed.

Measurement of electron flux in media bombarded by X-rays. R. D. BIRKHOFF, D. R. NELSON, H. H. HUBBELL, JR. and R. H. RITCHIE, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

Presently accepted procedures for the measurement of radiation exposure are based on the number of ions or electrons liberated by the passage of radiation through matter. The necessity for the introduction of R.B.E. values in order to characterize biological damage suggests that other factors may be involved. Among these is the distribution in energy of the liberated electrons as opposed to considerations of only the number of such electrons. Parallel plate ionization chambers have been constructed of graphite, aluminum and copper, and current-voltage curves for these have been obtained at effectively zero gas pressure when the plates are bombarded with approximately 3 r/sec of lightly filtered X-rays generated by a 250 KVCP machine. A saturation current is obtained at about 20 volts chamber potential, and this current corresponds to 2.3×10^{-4}, 11×10^{-4} and 63×10^{-4} esu/cm^{2}r for the graphite, aluminum and copper plates respectively. Under certain assumptions concerning the penetration of the surfaces by electrons, the energy distribution of the electrons inside the solid may be found from the first and second derivatives of the current-voltage curve. These distributions are observed only for those electrons which have at least enough kinetic energy (E_w) to escape from the solid (15.9 eV for Al and 11.4 eV for Cu). Inside the metal the flux of such electrons declines monotonically from the value at E_w and it falls to half value at $E_w + 05$ and finally reaches zero at $E_w + 20$.

A proposed emergency evacuation radioactivity monitoring system. J. M. GARNER, JR., Health Physics Division, Oak Ridge National Laboratory, Tennessee.

The desirable characteristics of an emergency evacuation monitoring system are discussed. The primary purpose of such a system is to supply information which will enable one to make decisions on when and how to direct evacuation from an area. The author believes that this can best be done with limited information from many locations rather than detailed information from a few points.

It was found that the existing air monitors at Oak Ridge National Laboratory could be modified to telemeter a portion of the needed information to a central location. To obtain information from many locations, a relatively inexpensive, remote beta–gamma monitoring system was developed. The remote monitor utilizes the average current

from halogen quenched G.M. tubes. The current from a G.M. tube is delivered to the central station over the same pair of telephone wires that delivers power to the remote unit. During a long term test the equipment was found to be reliable and require very little maintenance.

The proposed system will warn of a hazardous condition, and indicate its location, rate of spread and the approximate levels of the external and respiratory exposure hazards.

Session 2B

Health physics field operations. CHARLES L. WEAVER* and ROSCOE H. GOEKE.†

Health Physics in the support of field operations at the Nevada Test Site and Eniwetok Proving Ground is briefly discussed. At the test sites, the terms Radiological Safety and Health Physics are used interchangeably. Radiological Safety requirements necessary to control radiation exposure of all participating personnel at both sites are presented. The organizational setup for Health Physics support services and assignment of responsibilities for control of radiation exposure to personnel are discussed. The necessary supporting rad-safety functions, such as training of monitors, use of aircraft and instrumentation, are included.

* Radiological Safety Officer, Nevada and Pacific Operations Divisions, Ofc. of Test Opns., Albuquerque Operations Office, USAEC, Albuquerque.
† Radiological Safety Advisor, Office of Test Operations, Albuquerque Operations Office, USAEC, Albuquerque.

Large area decontamination. W. S. JOHNSON,* GORDON L. JACKS† and F. W. WILCOX.‡

In the course of a weapons testing program, the decontamination of large land areas is a continuing problem confronting health physics, scientific and construction personnel. These decontamination operations may be generally classed as follows:

(A) Those required for expediency during a test series.
(B) Those required before the start of a test series.

In the first case, the aim is to reduce contamination levels to something less than 100 mr/hr, while in the second, efforts are directed toward reducing the dose rate below 10 mr/hr.

Data has been obtained on the effectiveness of various decontamination techniques. Results are presented on the basis of personnel exposures, equipment used, radiological safety procedures required and the time expended on the operations to obtain the desired effect.

Our experience has shown that the nature of the contaminating event in many instances dictates the decontamination methods employed.

* Manager, Health and Safety Department, Reynolds Electrical & Engineering Co., Inc., Mercury, Nevada.
† Major, Signal Corps, U.S. Army, Radiological Safety Officer, JTF-7, TG-7.1.
‡ Supt., Radiological Safety Div., Health and Safety Dept., Reynolds Electrical & Engineering Co., Inc., Mercury, Nevada.

Electronic data processing as applied to personnel radiation dosimetry. W. S. JOHNSON,* GORDON L. JACKS† and F. W. WILCOX.‡

A brief discussion of the problems associated with automation in a legal dosimetry program is presented, including the various sources of error that occur in any program in which the series operations are conducted manually. A partial solution to the problem, a system employed by the on-site rad-safe organization for handling personnel radiation exposure data at the Nevada Test Site, is discussed.

A summary of personnel gamma radiation experience at the Nevada Test Site during 1957 test operations is presented with the following details of the film badge recording and reporting program:

(1) Electronic data processing machines utilized.
(2) Film badge issue technique.
(3) Recording methods.
(4) Report computation and distribution.

Economic aspects and other advantages of this system are discussed.

* Manager, Health and Safety Department, Reynolds Electrical & Engineering Co., Inc., Mercury, Nevada.
† Major, Signal Corps, U.S. Army, Radiological Safety Officer, JTF-7, TG-7.1.
‡ Supt., Radiological Safety Div., Health and Safety Dept., Reynolds Electrical & Engineering Co., Inc., Mercury, Nevada.

Electronic data processing and radiation exposure records. IRAL C. NELSON, Hanford Laboratories, General Electric Company, Richland, Washington.

A description is given of an application of electronic data processing methods to the maintenance of Radiation Exposure Records at the Hanford Atomic Products Operation. Described are: magnetic tape records, punched card inputs, computer dose determination, reports, relative costs and record storage. One year's successful operation has proven the worth of the application.

Automation of film badge processing and data reduction. MACK WILHELMSEN and DERAY PARKER, Instrument & Development Branch, Health and Safety Division, Idaho Operations Office, USAEC.

A complete system has been developed in which the film badge is identified and the identification and dosage data are automatically reduced to digital form, and presented on an IBM punched card.

To facilitate automatic processing, the identification is placed on the film badge in binary form instead of numerical. The individual's identification and area number is punched into a lead insert in the film badge as a series of small, round holes in binary code. After use, the film badge is passed through a machine in which the identification data is X-rayed onto the film insert in the same binary code. The film insert is removed from the badge, placed in a specially-designed tray and chemically processed and dried. The identification data is now a series of black, opaque spots and transparent areas on the film insert. The indicated dosage is represented by two

areas of variable density, dependent on quantity and type of radiation.

The tray of film is placed in a machine which automatically places each film into the film reader and returns it to the tray. In the film reader a light source illuminates the entire film. The identification number is obtained from the film by a series of photo-diodes corresponding to the black and transparent areas on the film insert. The photo-diodes are connected to a diode relay matrix in which the number is stored in binary decimal form, until read-out is completed by the IBM card punch.

The dosage data which are proportional to the density of the film, are obtained by two multiplier phototubes and associated electronic circuitry. A beta and gamma dosage reading is taken and stored in digital form, the beta dosage reading being corrected against the gamma component. All of the stored information is fed into an IBM card punch in serial form and punched into a data card. The reading process is completely automatic and is keyed to the cycle of the IBM card punch. The film is read at the rate of 15/minute. Several fail-safe circuits are incorporated to prevent erroneous information being recorded.

The use of statistical methods in health physics problems. M. M. SANDOMIRE, Staff Statistician, U.S. Naval Radiological Defence Laboratory, San Francisco.

General definitions of analytical statistical methods, emphasizing the statistical meaning of sampling, will be stated. The contribution of these methods to the solution of problems in the field of Health Physics will be given, with the use of some illustrative examples, such as the standardization of test methods, calibration of instruments, study of operator and instrument variability and bias and correlation of results obtained by different methods. These problems will, in turn, be related to the phases of monitoring procedures, that is assessment, delineation and control of a radiological situation, and the value of the principles of statistical design of experiment will be pointed out. The need for a probabilistic approach to the analysis of various types of operational problems and studies will be mentioned.

Reactor shield penetrations. R. L. FRENCH. Convair, A Division of General Dynamics Corporation, Fort Worth Division, Fort Worth, Texas.

Differential energy spectra in various media for point isotropic sources of monoenergetic gamma radiation have been obtained by a moments method solution of the Boltzmann transport equation. These data are utilized in a method which has been developed for calculating gamma spectra and dose rates from nuclear reactors. An elemental volume of the reactor core is considered as a point isotropic source of gamma radiation with a spectrum which is defined at several energy levels. After penetraton of each shield layer the modified spectrum is calculated and used as a source term for penetration of the next layer. For each iteration the calculation involves an integration over the source spectrum for each of the defined energy levels. After penetration of the final shield layer an integration is performed over the volume of the reactor core. Dose rate is obtained by applying the appropriate response function and performing an integration over the energy range. Results obtained with this method agree with BSF centerline dose rates, for example, within 10 per cent out to 120 cm in water and within 30 per cent out to 380 cm in water.

Session 3

Present practices in the estimation of internal dose. W. S. SNYDER, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

The central problem of Internal Dose estimation is that of determining the amount and distribution of the radionuclide in the body under various conditions of exposure. A compartments model assuming constant percentage of elimination usually requires a rather large number of compartments and thus a large amount of accurate data to evaluate the parameters used. It has the advantage of some inherent stability in making extrapolations over very long periods of time, but represents the body as a static organization. However, an empirical approach offers little guidance in estimating the effect of the data of conditions and periods of exposure which are markedly different from those of the experiment. The various parameters needed to determine the distribution in the body are briefly reviewed and the need of some general theory or indication of their dependence on chemical form or particle size, organ of deposition, species difference and individual differences is pointed out.

Low intensity spectrometry of the gamma radiation emitted by human beings.* C. E. MILLER and L. D. MARINELLI, Radiological Physics Division, Argonne National Laboratory.

Gamma ray spectroscopy has been developed at Argonne National Laboratory to identify and measure extremely small amounts of *in vivo* activity in humans. Techniques have been established which yield both whole body content and the location of the deposit. Equipment and facilities permit detection of as low as 2×10^{-10} curies of localized deposits.

These measurements are utilized to study the radioactivity of normal unexposed individuals, the long term effects of internal deposits and the transient behavior of materials after initial ingestion, inhalation, or from puncture type wounds.

The normally present *in vivo* radioactivity of twelve unexpected control subjects has been measured periodically since March 1955. Numerous individuals from various parts of the world have also been measured.

The long term effects (up to 35 years) are being studied with the co-operation of several hundred former radium dial painters and several thorotrast patients. The present body burdens of these individuals are correlated with clinical changes as detected by radiographic skeletal surveys. These studies will lead to a statistical evaluation of current permissible levels. In addition, since the Ra burdens vary over a range of as much as 500 to 1, the

presence or absence of a threshold effect may be established.

In the case of accidental intake, it is possible to determine (1) the amount initially taken into the body, (2) the initial distribution in the body (location in the upper lungs, bronchi, muscles, etc.) (3) the transport from initial location to other organs (4) the elimination rate by normal body functions or as influenced by chelating agents and (5) the fraction permanently retained by critical organs of the body.

All measurements are made in the Argonne Iron Room. The 8 inch steel walls shield out all environmental background. NaI(Tl) crystals are used for gamma ray detection because of their superior spectrometric properties. Whole body levels are measured by an $8'' \times 4''$ thick crystal utilizing either the 1 meter arc or the standard chair technique. Collimated crystals of various sizes are used to scan the individual and determine the distribution of activity.

The following cases, utilizing the above techniques illustrate particular uses which have been made of the gamma ray spectrometer:

A chemist, working with Am^{241} (a 60 keV gamma ray emitter) received a contaminated puncture type wound on his left hand. The wound was excised and chelating agents administered. Complete body scans shortly after the accident and two years later were made to determine the effectiveness of the chelating agents, the biological eliminating rate and the transport of the isotope from its point of entry to other parts of the body.

The body burdens of two individuals who accidentally inhaled Te^{129} as a powder were followed for 150 days. Measurements made using a variation of the one meter arc technique showed that the lung burden decreased by 50% every sixteen days. Data has also been obtained in inhalation cases for Pa^{233}, $RaSO_4$, Th^{227} and U^{235}.

The above cases prove the invaluable role that gamma ray spectroscopy plays in the understanding of *in vivo* behavior of radioactive material and the development of radiation therapy. The effectiveness and mechanism of chelating agents can be accurately followed. The behavior of the materials retained in the lungs and gradually absorbed by the blood can be studied and the excretion data curves explained. These techniques are also utilized in medical research in following tagged drugs and other materials.

* Work performed under the auspices of U.S.A.E.C.

Metabolic characteristics within a chemical family. PATRICIA W. DURBIN, University of California Radiation Laboratory and Division of Medical Physics, Berkeley.

A systematic study was made in adult rats of the biological behavior of high-specific-activity radioisotopes of 70 elements. Tracer quantities were injected intramuscularly in neutral isotonic saline or sodium citrate. Groups of rats were autopsied at various intervals, and tissues and excreta were assayed for radioactivity.

Distribution data was arranged according to the grouping of the periodic table of the elements. The anions including the halogens, the oxygenated or halogenated ions of Groups IV, V and VI and the transition metals were rapidly eliminated by the kidney. The monovalent alkali metals were distributed relatively uniformly in soft tissue with subsequent excretion by the kidney. The bivalent cations, except Hg, Cd and UO_2^{++} were primarily deposited in mineral bone and were eliminated slowly in both urine and feces. The tripositive elements of Group III, the lanthanides and the actinides were deposited in liver and bone apparently in association with protein, the dominant site depending on the relative stability of the protein complex. The liver fraction was eliminated in the bile without recirculation, while that deposited in bone was turned over at a slower rate than that of normal bone remodelling. The quadrivalent cations including Zr, Th and Pu were deposited almost exlusively in bone, and were held more strongly than the Group III elements.

The properties which appear to determine biological behavior are (a) the oxidation state stable at body pH, (b) the solubility of the stable state, (c) the tendency to be incorporated into organic compounds and (d) the tendency to associate with specific proteins.

Metabolism of radioactive isotopes which deposit mainly in the skeleton. BETSY J. STOVER.

Of the 102 known chemical elements more than half of them have been shown or can be predicted to deposit in the skeleton to a significant extent. These elements fall into three categories: the natural constituents of the skeleton, the elements which are chemically similar and those which have no physiological analogs and are generally characterized by low solubility at the physiological range of pH. Fortunately not all of these 50 to 60 elements are of importance in the problem of radiotoxicity. The natural heavy radioactive isotopes, plutonium, those fission products with high yields and relatively long half periods and those radioactive isotopes used extensively in research laboratories constitute the main problem.

The two isotopes Ra^{226} and Pu^{239} have been extensively studied, and they exemplify the chemically similar and the metabolically foreign groups of bone seeking isotopes, respectively. Their metabolism and average skeletal radiation doess will be compared. An additional problem results when the radioactive isotope in question decays to another radioactive isotope, for the metabolism of the daughter must also be determined for dosimetric calculations. The location of the parent, the recoil energy and the chemical identity of the daughter all appear to be important factors.

Consideration of inhalation hazards in the light of present practices in internal dose estimation. J. N. STANNARD, University of Rochester School of Medicine and Dentistry.

Current calculations of permissible body burdens and of limiting air concentrations for inhalation exposures involve three major sets of assumptions. These are: (1) the solubility of the compound in body fluids, (2) the choice of critical organ and (3) the fractional deposition and retention times in the lung. Current practices (NBS

Handbook 52 and ICRP 1955 edition) are to call the compound either "soluble" or "insoluble", the critical organ either lung or G.I. tract for "insoluble" materials, lung or organ of highest concentration for "soluble" materials, and to asssume the Harriman Conference model for pulmonary deposition and retention to be valid. This paper will take up these several assumptions in the light of current research on radioactive aerosols.

Pulmonary retention times have turned out to have no clear relation to the nominal "solubility" of the compound. Some very insoluble materials seem to leave the lung quite rapidly while some relatively "soluble" materials remain for surprisingly long times. The idealized terms "soluble" and "insoluble" seem to be quite meaningless in the biological system insofar as residence time in the lung is concerned. If readily soluble, material which leaves the lung quickly may appear in such organs as bone, kidney, muscle, etc. after parenteral administration. It may appear largely in G.I. tract, if it is not very soluble. Thus the distinction in terms of "solubility" is largely a basis for choice of critical organ after removal from the lung rather than probable residence time in the lung.

The circumstances under which a portion of the G.I. tract may be critical organ warrant close scrunity. Because of the considerable opportunity for radioactive materials to become incorporated with inert material the radiation dose to the gut wall and especially to the radiosensitive areas may be much lower than predicted by the total quantity of emitter present. Experimental studies with alpha emitters will be cited. Calculations based upon the assumption of G.I. tract as critical organ may turn out to be very conservative. Whether or not they are unduly conservative depends on the circumstances.

A major omission in current practices regarding critical organ is the neglect of lymph nodes. Data are now available which show that lymph nodes may easily exceed all other tissues in total amounts of material accumulated on a per gram basis. The factor is sometimes very large. The maximum exposure levels recalculated on the basis of lymph nodes as critical organ and their possible validity will be discussed.

The correctness of the assumptions in the lung model depends on the importance of particle size and the subsequent clearance rates and mechanisms. There is reason to think that the submicronic particle sizes will be of much greater importance in evaluating radioactive inhalation hazards than in ordinary dust exposures. Fortunately, the new data accumulating do not indicate wide discrepancy between the assumed 75 per cent deposition figure postulated by the model and the experimental values. This occurs despite predictions to the contrary on other bases. On the other hand, the rates of clearance from the lung vary so widely that the model becomes essentially a median. If any major decisions are involved, use of actual data for the compound in question is much more desirable. This is especially important since the errors are as likely to be on the hazardous as on the conservative side.

Thus current practices may require some modification. The data accumulated to date indicate strongly that there is no safe substitute for doing inhalation experiments and using actual experimental data wherever possible. Much of this new information is being compiled for this purpose and will be reviewed in part. Meanwhile, however, and for some time to come, we must expect to rely on the models as the only approach possible in many instances.

Dose to the intestinal tract due to ingested activity.* R. L. Hayes, M. M. Nold and C. L. Comar.

A technique has been developed for the measurement, in experimental animals, of the dose to the intestinal mucosa due to ingestion of unabsorbed beta emitters. The technique involves implantation of Schulman–Etzel small volume glass dosimeters in the wall of the gut and subsequent recovery after dosing. Extrapolation of the results has yielded a maximum permissible intake for yttrium-90 of approximately 17 microcuries for a dose of 0.3 rad to the lower large intestine. Calculation by others has indicated a permissible level of 3.2 microcuries. An indirect dose determination procedure with animals based on the observed intestinal content changes in yttrium-90 content with time was in excellent agreement with the dosimeter measurements. Further studies involving experimentally induced states of diarrhea and partial constipation have indicated the extreme importance of intestinal content flow rate as a factor in the dose received by the intestinal tract due to ingested activies. It is anticipated that portions of this study will be extended to include measurements in man.

* Work done under contract with the United States Atomic Energy Commission (Contract No. AT-40-1-Gen-33). Special contract with Armed Forces Special Weapons Project (SWPEF-5/927).

Metabolic interspecies correlations with particular reference to alkali metal ions. Chester R. Richmond and Wright Langham, Los Alamos Scientific Laboratory, University of California, Los Alamos, New Mexico.

Present methods of calculating maximum permissible concentrations of radionuclides in air and water require data on their absorption, deposition and retention (or excretion) by man. Human data for most nuclides, however, are extremely limited and use of available animal data is necessary. Extrapolation of animal data to man is usually specified as the basis of MPC calculations. However, direct substitution instead of extrapolation is actually employed because of the absence of interspecies correlations on which to base extrapolations. Using whole body *in vivo* counters, absorption, retention and excretion of isotopes of the alkali metal ions were studied in mice, rats, dogs monkeys, and man. Retention was followed until the body burden dropped to less than 1 per cent of the absorbed dose. In almost all species the retention curves of the various radionuclides could not be represented as single exponentials, but multiple rate equations were necessary to describe the data over the entire period of study. The retention of the alkali metal ions by the various species suggested a heterogonic or allometric relationship, i.e., $Y = aX^b$, in which Y was the time required to eliminate the first half of the absorbed dose ($BT_{1,2}$) and X was some parameter of body size (in this case the body surface area).

The measured $BT_{1,2}$ values for Na^{22}, Rb^{86}, and Cs^{134} or Cs^{137} in man were 11, 80 and 110 days, respectively. These values were in good agreement with values for man predicted from the heterogonic relationship. The $BT_{1,2}$ for K^{42} was estimated at 35 days. The biological half-times (T_b) for Na^{22}, Rb^{86}, and Cs^{137}, based on the major component of the retention curves, were 13.5, 98 and 143 days, respectively.

The sensitivity of whole body counters affords opportunity to collect limited experimental data on retention and excretion of radioisotopes in man. Their very short operation time and *in vivo* measurement capability make it possible to collect data from several species in search of interspecies correlations in radionuclide metabolism.

Session 4A

A new method of using ionization chambers for low level dose measurement. F. L. Rising, W. C. Roesch and R. C. McCall, Hanford Laboratories, General Electric Co., Richland, Washington.

A new method has been developed for the measurement of low levels of radiation using condenser ionization chambers such as the pencils used in radiation protection work. The method of reading utilizes the magnitude of the current pulse required to recharge the pencils after they have been exposed. Any charging voltage may be used without a change in sensitivity. Using a low charging voltage reduces leakage to a very low level. Full range of pencils charged to 20 volts is about 30 mr. Higher charging voltages can be used to increase this range. Sensitivity of a small fraction of an mr is available. Readings of 1 mr, 5.2 mr or 10 mr : 1 mr are possible at 95% confidence with a single pencil.

The system has been used at HAPO to measure background on the project, on the Columbia River and in the environs. Typical values are presented.

Other possible applications are accurate personnel monitoring and neutron dose measurement. This system lends itself to automation of records keeping.

A study of the response characteristics of various automatic air monitoring instruments. M. A. Greenfield, R. L. Koontz and J. J. Wooldridge, North American Aviation, Atomics International.

The response characteristics of continuous recording air monitors for various airborne contamination incidents have been calculated. The results are used to evaluate the relative merits of the following types: (1) Detector located directly underneath the collection system; (2) Detector located underneath collection system with a moving filter paper; (3) Detector removed a distance from the collection system with the filter paper passing from underneath the collector to the detection device. The effects of tape speed, simple exponential decay and fission product decay on the buildup of activity on the filter paper are shown. Formula and curves are given for the response of the detection system to pulse and step functions of room activity from which more complex cases can be derived.

A continuous air monitor for tritium. J. Brinkerhoff, C. A. Ziegler, R. Bersin and D. Chleck, Tracerlab Incorporated, Waltham, Massachusetts.

A continuous air monitor for tritium capable of giving an accurate indication of concentrations from 0.1 to 100 times the maximum permissible value (2×10^{-5} μc/cc) is described. The device is essentially a flow-type ionization chamber. By means of a novel mode of operation, however, its response is not subject to the ambiguity common to other devices of this nature.

Film characteristic curves produced by different ionizing radiations. R. Golden and E. Tochilin, U.S. Naval Radiological Defense Laboratory.

Characteristic curves for six different emulsions have been obtained with nuclear radiations of a wide range of specific ionization. Sources included alpha particles, protons, neutrons and beta, gamma- and X-rays. With the two most sensitive films (Eastman Type K and DuPont Type 555), identical characteristic curves were obtained for all sources. Curves for the remaining films were found to be dependent on the type of radiation. When log density was plotted against log exposure, the curves fell into two groups. For particles with specific ionization equal to or greater than that produced by 30 keV X-rays, the slope in the linear region approached 1.0. For beta-rays, high energy gamma-rays and electrons, slopes greater than one were obtained. In most cases identical saturation densities were observed. Measurements made over a wide range of radiation levels failed to reveal any presence of reciprocity law failure. Data is presented which shows that gamma-ray characteristic curves can be simulated with beta-ray exposures. This procedure can considerably simplify the running of a curve, particularly where high level exposures are involved. For the most sensitive emulsions such curves may also be obtained with low energy X-rays.

A dose rate dependent film dosimeter. Keran O'Brien, Leonard R. Solon and Wayne M. Lowder, United States Atomic Energy Commission, Health and Safety Laboratory, New York, New York.

A film dosimeter suitable for dose-rates from the region of 10 μr/hr to 100 μr/hr has been designed using a NaI(Tl) phosphor as a density amplifier. Because of the dosimeter's failure to follow the reciprocity law, the film density as a function of applied dose is complicated by a time dependence. The dosimeter is calibrated assuming an invariant dose-rate. The effects of some simple dose-rate variations are examined. It is found that for exposures of the film-scintillator system greater than 50 hours, a reliable estimate of the background dose may be obtained.

A film technique for measuring the exposure dose from plutonium. E. C. Watson, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

A review of the radiations from the various isotopes, coupled with results from step-wedge measurements, show

that the electromagnetic radiations from plutonium can be considered in three effective energy groups.

The first group consists of X-rays with an effective energy of about 17 keV. This radiation is identified with the return of orbital electrons to their normal energy state after internal conversion following alpha emission. The second group consists of gamma rays with an effective energy of about 60 keV. The third group consists of all gamma rays with energies greater than 300 keV.

With minimum shielding, the X-rays contribute about 65% of the total exposure dose. When measured outside present shielded hoods, this contribution decreases to about 30%.

The low penetrating power of these X-rays has been demonstrated, using a water phantom. Because of the ease with which they are absorbed, the critical organ is the testes, provided there is adequate shielding of the eyes. Their contribution to the absorbed dose is reduced to about 13% when absorption is considered.

The measurement of all three energy groups with monitoring film is complicated by the energy dependence of the photographic emulsion used. The energy dependence of duPont Type 502 emulsion is shown to peak in sensitivity at about 45 keV. At lower energies the sensitivity decreases very nearly exponentially and at higher energies the sensitivity decreases to a constant value for energies above 250 keV. The effect of shielding the film as in a film badge, improves the energy dependence but limits the minimum energy which can be measured to about 60 keV.

A description of the shielding features in the Hanford Film Badge and their use in measuring the exposure dose of the three energy groups is given. From a set of generalized equations, relating exposure dose and net density, expressions are derived for the dose due to each energy group.

Calibrating energies are obtained by use of a fluorescent X-ray source which produces essentially monochromatic X-rays.

* General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Fallout studies of surface contaminability and of the nature and distribution of activity in rainfall. F. P. Cowan, Health Physics Division, Brookhaven National Laboratory, New York.

Fallout measurements were made from May to October 1957 with the following objectives:

(1) To study the relative contamination of typical building surfaces by fallout from Nevada and the effect of decontamination procedures.

(2) To determine the amount and approximate composition of fallout in rain as a function of time.

Three sets of panels of six materials were exposed to the weather and counted periodically with a large flow counter probe. One set was scrubbed with water, one with detergent and one was unscrubbed. A number of interesting differences in behavior were noted.

An automatic sampler was constructed for collecting hourly samples of rainfall which were analyzed for gross activity, per cent particulates and percent of Mo^{99}, I^{131} and Sr^{90}. A three-channel low-level counter was utilized. The apparatus will be described and the results summarized.

A small personnel dosimeter for beta and gamma radiation. Andrew P. Hull, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

A dosimeter has been developed which may be worn conveniently anywhere on the body. The device is basically a parallel plate ionization chamber with closely spaced circular electrodes. The chamber has the same sensitive volume as the Victoreen pocket chamber and has an X-ray response independent of energy. The outer-ribbed structure is made of nylon and forms a short cylinder an inch and a half in diameter and three quarters of an inch high. Filling the space between the ribs is a conducting plastic film approximating the human epidermis in thickness. The electrical capacitance is not only in the region between the Aqua-dag coated plates, but also in the insulating plastic base between the lower plate and the metal cap which holds the dosimeter to the wrist or elsewhere. The total capacitance was made equal to that of the Victoreen pocket chamber by partially hollowing out the base and filling the resulting cavity with stryofoam. Beta calibrations were made with 2π geometry obtained by inserting the dosimeter in the paraboloidal cavity formed in a rotating solution of one of several beta-emitting isotopes used. The beta response is similar to that obtained previously for the Al and Mg chambers.

An ionization chamber laundry monitor. John S. Haneloser and John D. Chester, Brookhaven National Laboratory, New York.

The determination of the amount of contamination remaining on a garment after it has been washed is an important part of hot laundry operations. In the past garments have been monitored by measuring the contamination concentrated in the crotch with a GM tube probe. This type of spot check does not detect any isolated hot spots on other parts of the garment. To monitor the entire garment with a GM tube instrument is excessively time consuming for a large number of garments.

To overcome these difficulties a sensitive, large volume, ionization chamber has been constructed. Physically the chamber is rectangular in shape being 5′ high $2\frac{1}{2}$′ wide and 4″ deep. The center electrode is of a grid type construction mounted halfway between the front window and the back of the chamber. 180 volts is sufficient to saturate the chamber in a 0.5 mr/hr field. In order to insure β sensitivity the front window has an equivalent thickness of approximately 7 mg/cm^2. A line operated electrometer circuit is used as the measuring device and is equipped with an alarm circuit which may be set at the rejection limit for the type of garment being monitored. Two to three μc of liquid mixed fission products deposited on a garment give a full scale deflection on the most sensitive range.

Since the chamber monitors the entire garment the

results are independent of the location of the contamination. In practice, garments may be monitored at the rate of 7 per min while only 3 per min may be completely checked using a GM tube probe. Field tests indicate that this instrument is stable and trouble free. Background causes a meter deflection of about 20 divisions which is low enough to give reliable accuracy for monitoring garments.

Low cost, multi-purpose hand and foot monitor. J. W. Baum, D. W. Soldan and E. J. Smelter, Armour Research Foundation, Chicago 16, Illinois.

This paper describes a multi-purpose hand and foot monitor designed to meet the needs of a small to medium size research lab. It serves as a hand and foot monitor, area monitor, floor monitor and laundry monitor. Its special features include: audible and visual alarms, fast response, hand probe sensitive to alpha, beta, and gamma radiations, low cost (about $750.00) and reliability. Sensitivity of the unit is reported for various modes of operation.

Integrity of vinyl plastic suits in tritium atmospheres. H. L. Butler and R. W. Van Wyck, E. I. du Pont de Nemours & Co., Savannah River Plant, Aiken, South Carolina.

Tritium, unlike most radioactive materials, is readily absorbed through the skin. Clothing, generally accepted as adequate for protection from radioactive contamination, is of limited value when working with tritium. Air supplied vinyl plastic suits offer the maximum practical protection for such work. This paper describes the test, results and equipment used to determine the integrity of both one-piece and two-piece plastic suits in tritium atmospheres.

A portable calibrator for beta-gamma survey instruments using Sr^{90}. C. T. Nelson, North American Aviation, Atomics International.

A small, portable calibrator has been constructed for the calibration of portable beta-gamma survey instruments in a laboratory. This method uses the assumption that any beta–gamma detection instrument which has initially been calibrated with gamma can be maintained in calibration by comparing the initial instrument calibration from a gamma source with the reading obtained at various distances with Sr^{90}. The beta source is then used as a secondary standard. The entire assembly is easily shielded and thus the use of a costly radium or cobalt gamma source and calibration facility and the radiation exposures which result in calibration are miminized.

Intensities from less than 1 mrem per hour to over 10,000 mrem per hour are possible at the position of the instrument which is being calibrated. The desired intensity within this range is obtained by a turn of a knob. A large dial shows the indicated intensity and a separate scale is used for each type of instrument. The paper describes the electrical and mechanical systems, source sizes and cost of fabrication. etc.

A calibration facility for thermal neutron survey instruments. A. Wendell Carriker, Naval Research Laboratory, Washington, D.C.

Fast neutron survey instruments are commonly calibrated with standard neutron sources, such as Po–Be. However, it is a problem to calibrate thermal neutron survey instruments because of a lack of adequate facilities that provide fluxes in the range of interest. The vertical access port to the thermal column of the Naval Research Laboratory reactor has been modified for calibration of portable thermal neutron survey instruments. The main shield of this modified swimming pool reactor is equipped with a $9\frac{1}{2}' \times 4' \times 3\frac{1}{2}'$ graphite thermal column. The energy of nearly all the neutrons entering the access port from this column is less than the cadmium cutoff.

By insertion of shaped absorbers, the thermal neutron density has been made relatively uniform over the length of the BF_3 detectors in the survey instruments. With the reactor power at 100 KW, fluxes ranging from $\sim 1 \times 10^4$ nv to 2×10^2 nv are obtained by insertion of plastic sheets impregnated with varying amounts of boron oxide. The gamma level associated with these fluxes is less than 100 mr/hr.

A simplified air sampling method. Robert D. Thomas, Materials Laboratory, Wright Air Development Centre, Wright-Patterson Air Force Base, Ohio.

This report describes a fast and efficient method of determining the radioactive particulate matter content of air. A comparison is made with the previous time consuming method to show the advantages of the new method. The method is unique for calibration since the counting system can be easily calibrated for any isotope to be used for comparison purposes. This method is definitely an advancement in the state-of-the-art for air sampling techniques.

Session 4B

Probability of human contact and inhalation of particles. L. C. Schwendiman, Hanford Laboratories, General Electric Co., Richland, Washington.

Experiments were performed to determine the probability of particle contact and inhalation by individuals doing a variety of outdoor work. Fluorescent zinc sulfide of about 2 μ median diameter was used as tracer for the radioactive particulates of concern. Millipore filters were used to collect air samples adjacent to work or other operations performed in an environment containing a known concentration of the fluorescent particles. Sticky tapes placed at various locations on workers were later examined to determine particles contacting these locations. Probability of inhaling a single particle was determined to be 6×10^{-7} per hour for driving a vehicle containing a particle on the vehicle floor, 6×10^{-4} for sweeping a bus with a particle present on the floor, 3×10^{-5} for changing a tire with a loosely held particle, 2×10^{-4} to 1.5×10^{-2} per unit body area per particle per unit ground area for walking, digging and loading. The higher probabilities were obtained for lower portions of the body. Data

may be used to assist in estimating risk when particle deposition frequency is known.

Metabolism of inhaled radioactive particles.* D. H. Willard, L. A. Temple and W. J. Bair, Hanford Laboratories, General Electric Company, Richland, Washington.

Whole body turnover and tissue distribution were studied in mice for approximately 5 months after inhalation of insoluble radioactive particles, $Ru^{106}O_2$ and $Pu^{239}O_2$ (0.05 to 3 microns in diameter). The whole body burden was determined for the live mice with a whole body counter and at intervals after exposure groups of the mice were killed for radioanalyses of the tissue. Twenty per cent of the quantity inhaled was immediately exhaled and 23 per cent was deposited in the lungs. The remaining particles were deposited in the upper respiratory tract and subsequently swallowed. Of the 23 per cent deposited in the lungs half was cleared after one day. Although the whole body turnover rates of both Ru^{106} and Pu^{239} were best represented by power functions the estimates of the biological half-lives were obtained from semi-logarithmic plots of the data. From the seventh day after exposure the biological half-lives for Ru^{106} and Pu^{239} were approximately 30 days. The lungs contained over 70 per cent of the total body burden, the remainder being almost entirely distributed throughout the intestine, kidney, liver and spleen. These studies indicate that the critical organ for MPC considerations of $Pu^{239}O_2$ and $Ru^{106}O_2$ in air is the lung.

* Work performed under Contract No. W-31-109-Eng-52 between the Atomic Energy Commission and General Electric Company.

Metabolism of plutonium oxide after lung exposure.* W. J. Bair, Hanford Laboratories, General Electric Company, Richland, Washington.

Knowledge of the metabolism of plutonium oxide deposited in the lungs of animals provides a basis for predicting the behavior of plutonium oxide inhaled by humans. Because it is technically difficult to directly determine the whole body burden of plutonium it is important to establish relationships between clinical data and the total body or lung burden of plutonium. Six dogs were intratracheally administered 300 μg $Pu^{239}O_2$ particles (0.02 to 0.7 μc in diameter). Two dogs were killed 10 days after administration for determining the tissue content of plutonium. In others the quantity of plutonium appearing in the blood, urine and feces was determined for several months. The rate of excretion of plutonium in the urine and feces during the period 2 to 46 days after administration was represented by power functions. Of the small percent of administered plutonium appearing in tissues other than the lung at 10 days after administration the pulmonary lymph nodes, ovaries and bones showed relatively high concentrations. No relationship between the lung burden and the blood and urine content could be derived from the data.

* Work performed under Contract No. W-31-109-Eng-52 between the Atomic Energy Commission and General Electric Company.

Bremsstrahlung counting of Sr^{90} injected dogs. Charles W. Mays, Physics Group, Radiobiology Laboratory, University of Utah.

Strontium retention has been measured in Sr^{90} injected beagles by whole body bremsstrahlung counting. Fractional strontium retention may be expressed as $R = 0{\cdot}68\ t^{-0.21}$, where it is the number of days after injection, for beagles 17.7 months old at injection. This shows that in dogs, strontium retention is slightly less than radium retention. A method has been developed and experimentally verified by which low levels of Sr^{90} in a 10 kg dog can be determined with a standard deviation of less than 0.1 μc.

Evaluation of the hazard from inhaled insoluble oxides of thorium and uranium.* Robert G. Thomas, Ph.D. and Harold C. Hodge, Ph.D.

The fissionable nature of uranium dictates its primary use whereas thorium, although fissionable, is used mainly in the manufacture of alloys. The routine preparation of these elements for use creates a potential inhalation hazard, generally in the form of insoluble oxides.

The MAC's for insoluble thorium and uranium are identical and are based primarily on animal studies. This value (3×10^{-11} curies per liter of air) appears to be valid if calculated with lung as the critical organ. However, data accumulated over the past few years strongly indicate that pulmonary lymph nodes may be a better choice of critical organ.

The problem of assessing the present MAC's for thorium and uranium oxides requires more detailed information on the relative importance of these choices and the radiosensitivity of lung and lymph node tissue. The present status of this information will be summarized and the influence of variations in each parameter on the calculated MAC will be shown.

Urine analyses are a prime index of uranium exposure and appear suitable for estimation of lung burden. There is no adequate means of assessing a lung burden of thorium dioxide although thoron breath measurements, whole body counting and analysis of urine for daughter products present possibilities. Fecal analyses reflect an inhalation exposure to insoluble materials but the relationship is not quantitative.

* This paper is based on work performed under contract with the USAEC at the University of Rochester, Atomic Energy Project, Rochester, New York.

Effect of dietary calcium level on the chronic deposition of Sr^{90} and Ca^{45} in the mature rat. Ray F. Palmer, Roy C. Thompson and H. A. Kornberg, Hanford Laboratories, General Electric Co., Richland, Washington.

Experiments are described which measure the effects of time and dietary calcium level on the deposition and retention of chronically fed Ca^{45} and Sr^{90} in mature rats. Calcium content of the diet was varied between 0.03 and 2.0 per cent and the radioisotopes were fed for periods of up to 100 days. The ratio of Sr^{90}/Ca^{45} deposition in bone

was found to vary with both time and dietary calcium level. While deposition of both Sr^{90} and Ca^{45} was decreased by increasing the total calcium level of the diet, this effect was much greater in the case of Ca^{45} than in the case of Sr^{90}, and in the case of neither radioisotope was the effect of a linear one. These results bring into question the general practice of interpreting Sr^{90} hazards in terms of Sr^{90}/Ca ratios and supposedly constant discrimination factors, without consideration of the absolute Sr^{90} or inert calcium levels.

The relative effect of neutrons of 1.4-MeV and 14-MeV energies and gamma rays in the reduction of fertility in the male mouse. J. F. SPALDING, Los Alamos Scientific Laboratory, University of California, Los Alamos, New Mexico.

Male CF-1 mice were exposed to low weekly doses of Co-60 gamma rays, 14-MeV neutrons and fission neutrons over a 10-week period.

A complete record of the data from litters was maintained daily for 30 weeks to determine the efficiency of the two neutron energies relative to gamma rays in reducing fertility in the male mouse.

The sex ratio of mice born to animals in this study was 0.9 to 1.1 (males/females) and did not differ from controls, regardless of the type or dose of radiation received.

Fission neutrons were found to be about 6.5 times as effective in reducing fertility in the male mouse as gamma rays, and 14-MeV neutrons were approximately equal to gamma rays in their effect.

Complete sterility was observed in the high dose fission neutron group, but with the exception of one animal complete recovery was observed when the experiment was terminated.

Acute gastrointestinal radiation injury in rats following radioisotope ingestion. MAURICE F. SULLIVAN, PATRICIA L. HACKET, SIDNEY MARKS and ROY C. THOMPSON, Hanford Laboratories, General Electric Co., Richland, Washington.

Damage to the intestine of rats from an orally administered beta emitter, yttrium-91, is accompanied by hematologic changes similar to those resulting from exposure of the exteriorized intestine to external X-irradiation. However, the pathologic damage observed is markedly different. The small intestine exhibits a much greater sensitivity than the large intestine to ionizing radiations of external origin; but injury to the intestine from an ingested radioisotope is more closely related to the passage time for intestinal contents. The large intestine is severely damaged while the small intestine is relatively undamaged. Survival time after irradiation of the gastrointestinal tract from an internal source differs from that after external radiation by about the time required for passage of the radioisotope through the intestine.

Alpha radiation from orally administered plutonium-239 produces none of the symptoms associated with acute radiation damage to the intestine, such as are exhibited following Y^{91} administration. These results indicate that the relative effectiveness of these isotopes in causing injury to the intestine, when administered orally, bears no relationship to their biological effectiveness when given by other routes.

Session 5

Development of miniature chambers for low energy X-rays. R. GARRETT and J. S. LAUGHLIN, Department of Physics, Memorial Center, New York, N.Y.

The principles in the construction of small ionization chambers, particularly with regard to gas and wall materials and dimensions for use with X-rays in the 30 Kvp-120 Kvp X-ray energy region will be discussed. Details of the design of miniature chambers with conducting polystyrene walls and collectors of either polystyrene or aluminum will be presented. Characteristics of these chambers such as: dependence of sensitivity on energy, directional response and saturation properties will be described and compared with commercially available chambers.

Portable alpha survey instrumentation. HOWARD C. EBERLINE, Eberline Instrument Div. of R.E. & E. Co.

The problem of monitoring for alpha contaminants, either in the field or in the laboratory, has always been a nasty one. This is due to the fact that an alpha particle, although very energetic, has very little penetrating capability—5 milligrams per square centimeter in most cases being infinite. In order to admit the particle into any type of chamber means that the chamber must have an exceedingly thin wall. To make such a chamber durable for varying conditions for field and laboratory use requires a lot of thought.

We believe there are three basic types of alpha counters needed:

1. The simplest and most economical is for the laboratory where the environmental conditions are closely controlled. In this position we propose an air proportional counter with expendable chambers.
2. The second point of importance is the field or laboratory where environmental conditions are stringent but reasonable. Our solution to this is a gas flow alpha proportional counter where the probe economy is still maintained, but the atmosphere is supplied from a controlled source.
3. The third condition to be considered to where there are the greatest environmental difficulties, where extremes in temperatures as well as atmosphere are often encountered. We feel that the answer to this is a scintillation-type counter where a chamber can be made to withstand water immersion and tremendous pressure differentials.

This paper discusses in detail the functions, the advantages and disadvantages of the three types of instruments mentioned.

High-energy neutron dosimeter.* JOSEPH B. MCCASLIN, Radiation Laboratory, University of California, Berkeley, California.

Large plastic scintillators (1700 grams) have been employed to monitor high-energy neutrons and protons in the range 20 MeV to >400 MeV via the reactions $C^{12}(n, 2n)C^{11}$ and $C^{12}(p, pn)C^{11}$ respectively.

The convenient 20.4 min half-life, freedom from competing reactions, and carbon in the form of scintillator material all contribute to making this a desirable threshold detector of high sensitivity. One n/cm^2/sec will produce about 100 disintegrations per minute when the scintillator is irradiated to saturation. Without a cosmic-ray anti-coincidence shield the background activity is 1000 cpm, and work is presently being done to reduce this still further.

The scintillators were intercompared simultaneously with a bismuth fission counter in alternate proton-produced and deuteron-produced neutron beams at the Berkeley 184 inch cyclotron. These results will be given together with a brief description of counting procedures and equipment.

* Work done under the auspices of the U.S. Atomic Energy Commission.

Noble gas scintillation for neutron spectroscopy.* THEODORE M. JENKINS, Radiation Laboratory, University of California, Berkeley, California.

Two possible uses of a He noble-gas scintillation counter as a neutron spectrometer will be discussed. The first would utilize the reaction $He^3 + n - H^3 + p$, earlier used by R. Batchelor in a proportional counter for neutron energies up to 1.7 MeV. For spectroscopy at higher neutron energies, >10 MeV, one could use the scintillation produced by the He^4 nucleus recoiling from elastic neutron scattering. Not enough energy is imparted to the He^4 nucleus at neutron energies below ~10 MeV to make use of this interaction.

Work of other investigators in the field of noble-gas scintillation will be reviewed with mention of pulse resolutions and relative pulse heights from the different noble gases. Northrop and Nobles state that a mixture of 10% Xe and 90% He gives a pulse height half the size of that from a NaI crystal. Such a mixture would be of particular use in a He scintillator, as pure He alone gives a relatively poor pulse height. Nobles states that gas scintillators have the advantage of fast pulses and linearity of light production with energy loss in the gas, while Sayres and Wu mention the excellent γ-discriminations due to the low Z.

Construction of a gas-and-liquid-state noble-gas scintillation counter at Berkeley will be discussed. Problems such as the need for gas purity, and keeping the gas in liquid state during some experiments, were met and solved. Temperature and pressure effects were noted, as well as minimum alpha-particle energies that can be detected in such a counter. Finally, results of work done at Berkeley will be summarized, with special emphasis on neutron spectroscopy.

* Work done under the auspices of the U.S. Atomic Energy Commission.

Fast neutron spectrometry with a KI(Tl) crystal scintillation counter.* PAUL KIENLE,† Brookhaven National Laboratory and R. E. SEGEL,‡ Aeronautical Research Laboratory.

A fast neutron spectrometer utilizing the $K^{39}(n, p)A^{39}$ reaction has been constructed. The neutron energy spectrum is deduced by measuring the energy of the protons leading to the 1.24 MeV first excited state in A^{39}. The fast neutron induced charged particle reactions in K^{39} and the decay of the excited states in A^{39} have been previously investigated by other workers. A $\frac{1}{2}$ inch thick $1\frac{1}{2}$ inch diameter KI (Tl) crystal was irradiated with monoenergetic *d-d* neutrons produced by the BNL van de Graaff generator. The gamma rays were detected by a $3'' \times 3''$ NaI (Tl) crystal placed alongside of the KI crystal. The NaI crystal was shielded from the deuterium target, and the pulse height spectrum in the KI crystal in coincidence with 1.24 MeV gamma rays in the NaI crystal displayed on a 100 channel analyser.

For monoenergetic neutrons of 4.04 MeV energy the coincidence spectrum showed a single peak corresponding to 2.8 MeV protons. The full width of the peak at half maximum was 15% indicating a neutron energy resolution of about 11%. The ratio of peak height to background was at least 5 : 1 with an incident neutron flux of 15 neutrons/cm^2 sec. The yield was 4×10^{-3} proton pulses in the peak per incident neutron.

* Work performed under the auspices of the U.S. Atomic Energy Commission.
† Guest scientist from Technische Hochschule, Munchen, Germany.
‡ Guest scientist at Brookhaven National Laboratory.

Fast neutron surveys using indium-foil activation.* LLOYD D. STEPHENS, Radiation Laboratory, University of California, Berkeley, California.

Activation of indium foils by thermal neutrons has been applied to measurement of fast-neutron fluxes. Foils weighing 200 to 400 mg were placed in the centers of 3 inch radius paraffin spheres, which were in turn placed in boxes of cadmium. When the boxes were placed in the neutron flux the cadmium absorbs any thermal neutrons that may be present in the incident flux; the paraffin moderates the high-energy neutrons that penetrate the cadmium to thermal neutrons. The thermal-neutron flux is proportional to the incident fast-neutron flux over a range of about 20 keV to 20 MeV.

The foils are removed from the boxes and counted on a methane-flow proportional counter. When activated to saturation they give about 4 cpm per neutron per cm^2/sec. The counter has a background of about 10 cpm. The foils can therefore detect a flux of 3 to 4 neutrons per cm^2/sec.

High instantaneous neutron fluxes are easily detected

and counted by use of these foils. Many simultaneous measurements have been made easily by this method.

In measurements made at the Bevatron, the presence or absence of targets correlated very well with the foil data. Measurements also indicated much higher neutron fluxes in the vicinity of the tangent tanks than in the magnet quadrants.

* Work done under the auspices of the U.S. Atomic Energy Commission.

Neutron remote area monitoring system (RAMS).* CHARLES W. APPLEBY, University of California, Radiation Laboratory, Livermore Site.

A neutron RAMS has been designed and constructed that has provision for nine detector probes. Each probe consists of a moderated BF_3 counter and a transistorized amplifier. After further amplification the signal is coupled to a count rate meter through a pulse height selector. Telemetering of each detector probe is accomplished either manually or automatically. When operating automatically the recording time for each detector is accurately adjustable to any interval. The time lag between telemetering any two detectors is approximately one second. The time constant of the count rate meter can be adjusted to conform to the recording period of the detector.

Provision is made for a remote indicating meter, a remote alarm system and a strip chart recorder. The neutron RAMS gives instantaneous visual indication of the detector being monitored as well as automatically indicating on the strip chart the beginning of each detector sequence allowing comparison checks. The log-scale count rate meter is calibrated in units of MPL (maximum permissible limit) for neutron energies from 1 to 10 MeV. The calibration curve was constructed for a Ra–Be spectral energy distribution. The detection efficiency is 0.1 counts/sec/unit neutron flux. Range of the instrument is from 0.07 MPL to 127 MPL using $50n/cm^2/sec = 1$ MPL.

* Work was performed under auspices of the U.S. Atomic Energy Commission.

The use of light aircraft for radiation surveys following releases of large quantities of materials. D. M. DAVIS, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

The recent Windscale incident has emphasized the need for available light aircraft and proper instrumentation for aerial surveys following a release of large quantities of airborne radioactive materials. Early tests, 1951 and 1955, are reviewed and the program now in effect at the Laboratory is presented.

The flux and spectrum of cosmic-ray-produced neutrons as a function of altitude.* H. WADE PATTERSON, Radiation Laboratory, University of California, Berkeley, California.

A series of measurements of neutron flux as a function of altitude have been made with the following neutron detectors; bismuth fission ionization chamber, proton-recoil proportional counter and moderated and bare BF_3 proportional counter. In addition, data were also taken with a Simpson pile. Altitudes ranged from sea level to 40,000 feet. Appropriate treatment of the data can be made to yield information about the neutron spectrum, the average neutron energy and consequently the dose rate.

* Work done under the auspices of the U.S. Atomic Energy Commission.

ABSTRACTS OF PAPERS READ BY TITLE

Use of computers to evaluate internal dose.* BIRNEY R. FISH, Health Physics Division, Oak Ridge National Laboratory, Tennessee.

In every phase of internal dosimetry, from the experimental study of how a radionuclide is distributed in the body to the interpretation of excretion or body burden data, there are tasks that may be made easier by using available computers. Distribution and excretion models may be tested by means of an analog simulator and the same device may be used to aid in designing the general course of the experiment. The experimental data may be handled with greater ease and accuracy by using automatic data processing methods. Statistical evaluation of the data, including fitting curves to the experimental points, can be greatly facilitated, especially where there are large quantities of data.

There is a method proposed by Bernard for estimating lung burden of a poorly soluble compound from a polynominal fit of the excretion data. That procedure may be readily carried out on a digital computer. An alternate method for estimating lung burden by means of an analog simulator has been applied to an actual exposure and compared favourably to the results of external gamma ray measurement. The analog simulator also provides a simultaneous estimate of other organ burdens. A combination of the two methods would seem to offer the greatest advantage.

* Work performed in the Health Physics Division, Oak Ridge National Laboratory operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

Radiation safety monitoring of reactor demolition. GERALD T. LONERGAN, Argonne National Laboratory, Illinois.

A chronological history of the Radiation Safety Program employed during the decontamination and dismantling of CP-2-3-3′.

The reactors, the first to be dismantled in such a manner, were located at the Palos Park Site of Argonne National Laboratory. The discussion of this unique operation provides information of the planning and preparations prior to actual initiation of decontamination operations. A description of the physical extraction, chopping and methods of handling of the reactor fuel is discussed from a Radiation Safety viewpoint. A discussion of the actual method of disposal of the reactor shells is included.

Radioactive decontamination of the 60 ton overhead crane in a separations facility. D. F. KOBERG, Hanford Laboratories, General Electric Co., Richland, Washington.

Decontamination of the overhead crane was concentrated in a nine day program. Crane catwalk time limits which prior to cleaning, ranged from less than 30 seconds to six minutes, were increased to 10 to 20 minutes for 50 mrems exposure.

The decontamination was accomplished through the use of an organic solvent (methyl chloroform) air blown on to the contaminated surfaces through a manually directed half inch spray nozzle. The nozzle, held as close as possible to the contaminated surface was especially efficient for cleaning cracks, crevices and eroding contaminated paint, oil and grease.

Radiation exposure control was maintained through the use of closed circuit television and manually operated, electronic timekeeping devices called Dosage Rate Integrators.

The ten day program represented the work of some 110 individuals who expended a total of 10.5 rems of radiation exposure to accomplish the work.

Development of an environmental monitoring program for the Curtiss-Wright research reactor and nuclear facility. JOHN L. DONOVAN and CHARLES E. ROESSLER, Health Physics Section Nuclear Power Department, Curtiss-Wright Corporation, Quehanna, Pennsylvania.

The Curtiss-Wright Corporation has constructed the first privately owned Research Reactor and adjoining Nuclear Facility at its Quehanna site, an 80 square mile tract of land in central Pennsylvania. At present the facilities include a one megawatt swimming pool type reactor, a radioactive materials laboratory and a radioactive waste handling plant.

This paper is a presentation of the development of a practical environmental monitoring program for this area. Items to be considered include the topography and meteorology of the location, the influence of low population density, the effect of state and federal regulations and the anticipated activities at the site.

To date the program has included radioactive assay of subsurface and surface waters, vegetation, soil, air, fallout and rainwater. A discussion of the factors involved in re-evaluation of the program after one year is included.

The influence of fallout on an environmental monitoring program is pointed out. The presence of this fallout activity promotes a discussion of the problems associated with conforming to state and federal regulations governing activity in air and water. Also mentioned is the influence of the greater assumption of regulatory powers by the states.

The radiological research program of the FCDA. JACK C. GREENE, FCDA.

Reduction of effects on people of nuclear radiation produced by an enemy attack may be accomplished in two ways: (1) by reducing the exposures, or (2) by reducing the biological damage produced by such exposures. The approach intended at reducing direct exposure has a higher anticipated pay-off, although continued study of the biological effects of radiation and methods of reducing, or repairing, biological damage must also be emphasized.

As a basis for studying the physical as well as biomedical protection problems, expansion of knowledge relating to physical and chemical characteristics of fallout material as well as characteristics of initial and residual radiation is necessary to determine such things as: (a) the practicability and efficiency of decontamination and the various means and agents that might be used; (b) the solubility of the fallout material in water supplies, and other radioecological problems; (c) the energy spectrum as it varies with time; (d) the penetrability of various energy radiations in various building materials and various geometries.

Additional information regarding distribution of the radioactive material in the nuclear cloud, along with the effects of weather on fallout deposition and micro-meteorological influences such as the effects of uneven terrain and high mountain ridges, the effects a mass fire might have on fallout deposition, etc., is necessary to predict where the fallout will be deposited.

After theory is developed and experimentally tested it must be applied to practical situations. It is necessary to have a means for evaluating and exploiting the available shelter in existing buildings which are constructed of different materials and differ in size, shape, and height. Additional fallout shelter requirements depend on this information.

The enormous problems of reclamation and rehabilitation of contaminated areas must be studied. A shelter program, as necessary as it is to the survival of a significant percentage of our people under conditions of a heavy enemy attack, is not enough. Means must be devised to assure continued survival. People cannot live indefinitely in shelters or in their basements on existing food and water supplies. Eventually normal activities must be resumed and agricultural and industrial production restored.

The FCDA Research Programme intended to accomplish these objectives is explained in the paper.

Radiological protective apparel. T. C. MEHAS, Hanford Atomic Products Operation, General Electric Company, Richland, Washington.

This presentation establishes the need for systematic concentrated effort directed to the development of

radiological protective apparel. Notice is made of the unavailability of and necessity for apparel designed specifically for radiological applications. It outlines the basic deterrents to successful development and application and further clarifies the many sub-surface factors affecting acquisition of appropriate apparel. Special emphasis is given to the dependence of atomic installations on commercial manufacturing organizations for supply of essential apparel. Factors influencing the rapport of these concerns are delineated.

Hanford Atomic Products Operation's approach to establishment of effective practices is presented, along with a report on the progress which has been achieved and on goals for the future. Resultant benefits are reflected in increased economy, protection and personnel morale.

In consideration of the basic similarities throughout the industry, Hanford's experience should find wide application and should tend to stimulate interest and activity in a field too long neglected.

Radiation dose received by passengers and crew on plane carrying the maximum number of radiation units. D. M. Davis and E. D. Gupton, Oak Ridge National Laboratory, Oak Ridge.

Official Air Transport Restricted Articles Tariff No. 6A contains the rules and regulations governing the transportation of radioactive materials by air. As the Laboratory is a major contributor to air shipments of radioactive materials, it was deemed advisable to make some direct measurements to determine radiation exposure to personnel in aircraft carrying the maximum number of radiation units. From measurements made it was concluded (1) that the maximum exposure would not exceed 10 mr/hr if all regulations are enforced, (2) that regulations would limit some types of planes to a maximum of 5 units and (3) minor changes in regulations are in order.

Beta field dose rate corrections for portable dose rate instruments. J. F. Evans and I. C. Nelson, Hanford Atomic Products Operation, General Electric Company, Richland, Washington.

Beta radiation measurements taken with many portable dose rate instruments having a window size less than the source measured are in error due to wall shielding.

An empirical method is described which will provide corrections to portable dose rate instruments used in an extended beta radiation field. Graphs showing corrections for various source sizes and source-instrument distances are given for the Cutie Pie, Juno and Soup Bowl, dose rate meters.

Metabolism of Zn^{65} in the rat. John E. Ballou, Hanford Laboratories, General Electric Co., Richland, Washington.

Studies of Zn^{65} metabolism were undertaken as part of a general investigation of the radioisotopes present in effluent from the Hanford reactors. Following intravenous administration Zn^{65} was excreted rapidly largely via the feces. Immediately after injection the highest concentrations of the radioisotope were deposited in the visceral organs especially in the liver. Retention components with half lives longer than 35 days could not be demonstrated for most soft tissues. Initially low concentrations of Zn^{65} deposited in bone and pelt were retained much more tenaciously. After 10 days the concentration in bone was higher than in any other tissues and analysed. The retention curve for bone indicates a rather uniform exponential decrease in concentration with an effective half life of approximately 120 days. Applying this figure to the MPC calculation decreases the maximum permissible concentration of Zn^{65} in water from 6×10^{-2} μc/ml (ICRP value) to approximately 1×10^{-2} μc/ml. Studies are presented showing the effect of carrier concentration and the influence of plant and animal materials on the gastrointestinal absorption of Zn^{65}.

Is Health Physics a profession? A. A. Jarrett, North American Aviation Atomics International.

One of the most fundamental problems in Health Physics can be illustrated by the question frequently raised by technical personnel outside the field of Health Physics, "If Jones is such a good scientist, what is he doing in Health Physics?" An associated problem concerns itself with the evaluation by salary administrators in the low rating of jobs performed in operational radiological safety. In view of this, the shortage of trained technical personnel, and the growth of the atomic energy program is not this the time for an "agonizing reappraisal" of the potentialities and limitations in Health Physics?

The training program for health physicists at Atomics International. J. C. Lang, North American Aviation Atomics International.

The increasing use of radioactive materials and nuclear reactors has emphasized the need for trained specialists in the field of radiological safety. To meet this need at Atomics International—a Division of North American Aviation, Inc.—a formal training course was established to supplement on-the-job training. The course is intended to provide an analytic foundation for the practical solution of radiation protection problems as they occur in day-to-day operations.

This paper will describe the course, evaluate its results and indicate the direction to follow if the needs of the growing atomic energy program are to be met.

Prediction of radioactive disposal problems in a typical organic cooled and moderated power reactor. A. R. Piccot, North American Aviation Atomics International.

Methods of calculating expected activity from the organic coolant, impurities and corrosion are described. Activity for a typical reactor is presented. Leakage and removal rates of contaminated liquids and gases from the system are assumed and related to MPC values. Processing or dilution requirements and methods to permit disposal or release to the environment are described.

Basic technical steps to standards for radiation protection. MASASHI SUZUKI, M.D., M.P.H. Institute of Industrial Health, Ministry of Labour, Japan.

The history of Radiological Protection shows the establishment of some standards for the protection was quite important for practices. The most essential one is the settlement of Maximum Permissible Levels for Radiations. The concept of the item goes back to 1902, but the earliest quantitative suggestion was made by Mutscheller in 1925. Since then, numerous efforts were poured to the problem nationally as well as internationally. For the external hazard, step-by-step progresses were taken. The first international decision was made in 1934 in Zurich and 0.2 r/day as a tolerance dose. Important change was promulgated by U.S. Advisory Com. in 1936. The standard, 0.1 r/day, was an aim during World War II.

The most important, far-reaching determination was made by the Nat. Com. on Rad. Protection of U.S. in 1948, and lowered the level to 0.3 r/week. Considerations were given to the critical organ, age, emergency dose and RBE. Chalk River Conf. and Intl. Congress of Radiology in 1950 made further steps for the effort, and important international decisions were made in 1953 in the Recommendation of I.C.R.P. Many important considerations were taken for the standard, such as, approval of new unit "rad", definition of "permissible", determination of RBE, etc.

U.S. Nat. Com. of Rad. Protection issued the recommendation in 1954, and added some extending concepts such as weekly dose fractuation, non-continuous exposure, radiation tolerance status. After these recommend., greater weight was put on accumulated doses of radiation relating to genetic effect and long-term chronic effect on the population. More recommendations were issued by Brit. Res. Council and U.S. Ntl. Acad. of Science. 200 rem and 10 rem for individual per lifetime respectively, according to those working occupationally and living in the population, were recommended in I.C.R.P. in 1956. However, we have much difficulty of supervising the standard of accumulated doses for individuals in practices at present. Also more intensive efforts should be taken for collecting data and assessment of the effect of long-term, low-dosage radiation. More extending investigations should be made for factors of accumulation of radiation, of time-dose rate fractionation, of biological variations of resistance to radiation, conditions influencing to the resistance, natural background of radiation, genetic effects, etc.

As for the Permissible Standards for internal hazards, the problem of Ra poisoning was reviewed, which was the first technical step for the standard for internal hazards. 0.1 μc of Ra was taken as standard for Manhattan Project, but since the situation was drastically changed after the war, the recommendation of allowable limits for internal radiations was promulgated in 1949 at Chalk River. This was followed by the Recommendation of I.C.R.P. in 1953. Two basic criteria for the permissible body burden were taken; comparison with 0.1 μc of Ra, and 300 mrem/week of exposure to critical organs. Numerous figures were calculated by the criteria for various radioactive substances to assess the MPL for internal hazards. However, we are standing on too many assumptions whose scientific bases and stage of knowledges are quite ambiguous and uncertain. The author criticized the validities of these assumptions from biological viewpoint, and emphasize more directed and concentrated efforts should be taken to make these assumptions more certain and reasonable, though tremendous efforts might be necessary for it. Since it is felt these recommended figures are arbitrary at present, and as necessary steps are so numerous, the all-round policy should be avoided, and well-organized international group of experts is desired to direct the investigations and assessments of these listed values from time to time, in order to get more adequate and reasonable standards.

As the conclusion, since social demands for more adequate and practical standards of MPL for radiation are becoming greater and stronger, while we are still in a half way to pave for technical steps to it, I hope New International Steps could be taken to discuss the problem, to direct investigations for it, to assess data more effectively and authoritatively, to collect data rapidly and to exchange opinions of experts in every nation.

A scope of the science of radiological protection. MASASHI SUZUKI, M.D., M.P.H. Institute of Industrial Health, Ministry of Labour, Japan.

Since Radiological Protection is one of the essential necessities for the utilization of atomic energy, much effort has been made for the development of science and techniques of the field until now.

Many experts of the field called the science "Health Physics" and discuss the scope of it. Parker, Cowan, Anderson, and Claus, discussed the problem and attempted to define the scope of the field as well as described the relationship of the science with other numerous neighbor areas. Some experts incline to use the name "Radiological Protection" or "Radiological Health", because the actual aim of the science is to establish the protection of human life and health against radiation hazards. According to recent development of the science, I feel the name "Health Physics" is becoming rather inadequate, though the physics is still one of very important components of the science. Science of "Radiological Protection" would be more adequate and reasonable, because the field should inevitably include many different studies from physics.

It is always explained that radiological protection is standing on border lines of many different sciences and technologies, and some complain such sciences lying in border lines would be difficult to be established as one. However, the aim of the science is quite clear and only one which is the protection of human life and health from radiation hazards, so that it should be unified and consolidated.

Since any area of science includes some borderlines of other sciences and any borderline is artificial is its nature there must be no hesitation to eliminate such artificial borders, if the aim of the science could be unified. There are lots of evidence that the borderline of science is always changeable in the history of science.

Though the Science of Radiological Protection involves

many different areas, three main fields are most important key-stuffs of the science: namely, radiation health physics and engineering, biological and medical radiation health and environmental radiological protection.

The following scope is proposed by the author as the schema of it.

SL		Study of safe equipments and operations to radiation.
CO		Study of radiation shielding and protective design.
IG	Radiation Health	Study of radiation health engineering.
EI	Physics & Engineering	Study of radiation health dosimetry, and instrumentation.
NC		Study of radiation health chemistry.
CA		Study of radiation hazards evaluation.
EL		Study of radioactive waste disposal.
		Study of environmental radioactive contamination.
OP	Environmental	Study of radiological ecology.
FR	Radiological	Study of environmental health
O	Protection	protection from radiation.
		Study of public and civil protection against radiation.
RT	Biological & Medical	Study of radiation biology for health protection.
AE	Radiation Health	Study of radiological health supervision.
DC		Study of radiation injury treatment. (incl. diseases).
IT		Study of radiological preventive medicine.
OI		Study of radiation eugenics.
O		Study of radiation safety psychology.
N	Miscellaneous Fields	Study of education for radiological protection.
		Study of legal control and administration for Rad. Prot.
		Study of compensation and security for Rad. Prot.
		Study of radiation protection statistics and recording.

The weakest point of the science at present lies in Environmental Radiological Protection. There are many unbalanced developments in the science and much effort should be poured to biological and medical fields for reasonable radiation hazards evaluation. More studies are sincerely recommended for radiological ecology, eugenics and preventive medicine. The weight has been put on the first field too strongly until now, and I am afraid unbalanced developments will break its real effective achievement for the true purpose in the future. More emphasis should be made for reasonable consolidation of the science and effective co-operation among each area of the science.

Protective clothing (Film). E. I. du Pont de Nemours & Co., Inc., Aiken, South Carolina.

We recently completed at the Savannah River Plant a motion picture "Protective Clothing". This 25 min. color film was made primarily to train radiation workers in the proper techniques of donning and removing protective garments. Analyses of our skin contamination records showed that personnel unskilled in these techniques are likely to contaminate their skin. Other plants and laboratories may have similar training films of particular interest to health physicists. We suggest that the program committee for the Berkeley meeting consider such films as part of the program. We would be specially interested in films from other sites that might be of value to us in our training programs and probably other locations would have interest in our effort.

Health Physics Pergamon Press 1958. Vol. 1, pp. 235–254

THE SECOND UNITED NATIONS INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY GENEVA 1–14 SEPTEMBER 1958

A SELECTION OF PAPERS AND TITLES OF INTEREST TO HEALTH PHYSICISTS

P/100. The contamination of milk with radiostrontium from fall-out. G. W. Cox, A. Morgan and R. S. Tayler (United Kingdom).

1. The diet and milk from a herd of cows have been analysed for calcium and strontium-90 at regular intervals for over a year. The relationship between the mean strontium-90/calcium ratio in the diet and that in the corresponding milk was studied, and the discrimination against strontium observed by previous workers verified. The crops contributing most strontium-90 to the diet were identified and the reasons for the seasonal variation in milk activities clarified.

P/171. The determination of Sr^{90} and other β-emitters in human beings from external measurements of the bremsstrahlung. Kurt Lidén (Sweden).

Absolute scintillation spectrometer determinations of the internal and external bremsstrahlung from pure β-emitters such as P^{32} and S^{35} can be used for a calculation of the total number of photons produced in a certain energy interval. If the β-rays from the source are completely stopped in a radiator, the figures in the table below are valid for P^{32}.

Energy interval	Number of photons per disintegration, per cent Radiator material			
keV	C	Al	Fe	Pb
20–200	2.1	2.9	3.5	3.2
200–1000	0.4	0.7	1.3	3.8

According to these results it is obvious that a radioactivity assay based on bremsstrahlung measurements may be performed with the scintillation spectrometer technique. Small β-sources with a fixed geometry have been routinely assayed during the last years in several laboratories. Large irregular sources such as human beings and animals are more difficult to handle. As most of the photons in the continuous bremsstrahlung spectrum have low energies, the self-absorption and scattering in a large source, as well as the size and shielding of the detector, have to be considered carefully.

For the past five years the author has studied the production of bremsstrahlung and its possible use for *in vivo* localization of β-emitters in patients and the determination of the total body activity of β-emitters such as P^{32} and Sr^{90}. During a visit to the USA in 1957 it was possible to extend the investigation in this field at the Radiology Department of the University of Pennsylvania, Philadelphia and at the Health Research Laboratory, Los Alamos.

Measurements on patients treated or tested with P^{32}-labelled compounds have shown that the gross distribution of the tracer in the body can be followed by manual scanning and that it is possible to detect a significant uptake of P^{32} in small sections of the skeleton and in organs where metastatic lesions are suspected.

The study of the possibility of quantitative determination of small amounts of Sr^{90} in human beings has been performed with P^{32} as a test isotope and these results have then been compared with Sr^{90} measurements in phantoms. The experimental data show that the most efficient arrangement of the detector in relation to the background requires the use of a thin NaI (Tl) crystal, 6–10 mm thick. Measurements, in the iron-shielded room of the Health Research Laboratory at Los Alamos, on a person containing 2 μc of P^{32} showed that 0.2 to 0.5 μc of P^{32} should be detectable. A large crystal, having a diameter of 20 cm and a rather unfavourable thickness of 10 cm, was used. The geometrical efficiency was only about two per cent. The use of several thin-crystal-detectors around the body increases the geometrical efficiency considerably. The lower limit of the external *in vivo* determination of Sr^{90} thus lies in the interval between 0.1 μc and 0.01 μc.

The precision of the measurements depends also upon the accuracy with which the normal γ-spectrum from the human body can be determined, particularly the low-energy end of this spectrum. This problem is being further investigated.

P/177. Studies on soil-plant—animal interrelationships with respect to fission products. Lars Fredriksson, Bengt Eriksson, Bertil Rasmuson, Bo Gahne, Kay Edvarson and Kerstin Low (Sweden).

The main health hazard from low-level release of reactor-wastes of different kinds is probably due to the incorporation of certain fission products in food-chains.

An evaluation of these hazards necessitates rather detailed studies of the behavior of the different fission products in the food-chains characteristic for a certain milieu as it is to be expected, that appreciable differences exist, due to climatological conditions, soil type etc. especially in the first links of the chains. For some years a joint program for such studies has been established in Sweden and in the following some of these will be presented.

The influence of carrier calcium and potassium on the uptake of Sr^{90} and Cs^{137} from soils by different plant species

1. The influence of liming on the uptake of Sr^{90} by plants from a soil with very low natural content of exchangeable Ca has been studied in pot experiments. 5 lime levels and 16 important agricultural and horticultural plants were used. In 3 species—red clover, mustard and buckwheat—the experiments included both diploid and tetraploid plant forms.

Depending on plant species liming decreased the Sr^{90}-uptake with a factor of 3 to 10. Calculated on dry matter basis deploid plant forms took up definitely more Sr^{90} than did tetraploid forms of the same plant species.

2. Increasing amounts of carrier-Sr had but little influence on the Sr^{90} uptake.

3. Using 9 widely different soils the influence of increasing amounts of carrier-Cs and potassium on uptake of Cs^{137} was investigated. The uptake of carrier free Cs^{137} was on the whole very low, though it varied with soil texture—and apparently with clay mineral type and state in soil—and content of exchangeable K in the soil. Depending on soil type adding of Ca-carrier *increased* the uptake of Cs^{137} with a factor of 30–100 and the *total* uptake of Cs with a factor 10^5–10^6. Potassium decreased Cs^{137} uptake with a factor of 4–16 depending on soil type.

4. The uptake of Cs^{137} by 6 different plant species from 3 out of the 9 different soils has also been studied. The analytical work on the harvest products from these experiments has, however, not yet been finished. All experiments are followed up with extensive soil and plant analyses.

Soil-plant-animal interrelationships with respect to the absorption of Sr^{90}

In one of the experimental series mentioned above diploid and tetraploid red clover strains of the Resistenta type were cultivated on different Ca-levels. In all experiments the diploid has incorporated significantly more Ca and Sr^{90} than the tetraploid although the two strains are genetically identical except for the chromosome doubling. The average differences calculated on the dry weight between these two strains were: Ca 13.2% and Sr^{90} 15.5%. The cause of this difference is not known, but some nutritional experiments with mice were started in relation to this problem. The purpose was partly to study the quantitative incorporation of radiostrontium in mice, fed with the two strains and partly to investigate if they can discriminate Sr^{90} in relation to the actual Ca-content in the plants.

Except the Ca-Sr^{90}-content in the plant material the feeding ration must be absolutely Ca-free. Our synthetic feeding ration, supposed to be nutritionally complete, consisted of clover 20%, casein 20%, dextrin 30%, sucrose 20%, vegetable fats 8%, minerals 2% and vitamins. A certain amount of Ca^{45} was added to this mixture.

Growing mice—individually fed—were used and during the whole experiment feces and urine were collected separately every day. In the chemical analyses Ca and Sr were separated with a paperchromatographic method, and tested to give quantitative results. These analyses have not yet been finished.

When the feeding experiment was finished the animals were killed and used for activity measurements in the skeleton. So far the results suggest a higher radioactivity in the skeleton of the mice which were fed with diploid clover. The use of diploid clover thus seems to involve greater risks for the distribution of radioactive Sr.

Transport of long-lived fission products in Swedish soils

In the autumn of 1956 a study of the transport properties of long-lived fission products under field conditions was started. At five different places in Sweden test areas were selected which represented different soil types and climatological conditions. The test areas had the dimensions 2.5×2 m^2 and were placed on pasture-ground, not cultivated for some years. They were sprayed with a carrier-free solution of Sr^{90} (0.1 mc/m^2), Ru^{106} (1mc/m^2), Cs^{137} (1 mc/m^2) and Ce^{144} (1 mc/m^2). The first samples were taken about 1 month after the spraying and after that samples are taken each spring and autumn. Ru^{106}, Cs^{137} and Ce^{144} are determined by γ-spectroscopy directly on the samples. Sr^{90} is determined after chemical separation. The concentration of the different nuclides is determined as a function of the depth after which the transport properties for the different nuclides is calculated and correlated to the soil type and climatological conditions.

P/220. On the excretion of strontium-90 and cesium-137 by the human. C. G. Stewart, A. J. W. Hitchman and N. Jupe (Canada).

1. The radiation dose received by an individual who has an internally deposited radioisotope is usually deduced from radiochemical determination of the rate of radioisotope excretion at finite intervals after the time of exposure. In principle, the determination of dose is direct if an equation can be obtained relating excretion rate to time and if, from this, another equation may be derived relating body burden of retained radioisotopes to time. The integral of the latter equation over a chosen interval of time, multiplied by suitable coefficients to take account of the energy of decay, and divided by the mass of tissue in which the decay takes place, allows calculation of the dose in rads received by the tissue.

2. Experimental evidence already obtained with uranium, plutonium and radium indicates the excretion rate may be expressed, more precisely than by the commonly used exponential model, with an equation of the form $y = at^{-c}$ where y is the amount of radioactivity excreted per day, t has the usual meaning of time, and c is a parameter

characteristic only of the specific isotope. To assess the total radiation dose received within life expectancy from the rate of excretion of other radioisotopes it is important to determine whether their excretion rates can be expressed by a similar relation, and to determine characteristic values of c.

3. The rate of excretion of Sr-90 from two accidentally contaminated persons has been found to follow the relations,

(1) $$y = 20{,}600\, t^{-12.0}$$

(2) $$y = 8200\, t^{-1.10}$$

where y in each case is the daily excretion in urine of Sr^{-90} expressed in $\mu\mu c$ at t days, and where the two cases were followed experimentally for 490 and 734 days respectively. Similarly the rate of excretion of Cs-137 by one human has been found to follow the relation

$$y = 4080\, t^{-0.53}$$

for 386 days.

4. The implications of an exponent with an absolute value greater than unity in the excretion-rate equation of Sr-90 and less than unity in the excretion-rate equation of Cs-137 are discussed in terms of evaluating the total radiation dose received by an individual within life expectancy. The cooperation of many laboratories in the collection of excretion-rate equations for radioisotopes would make it possible to obtain truly representative values of the parameter c in humans and permit accurate radiation-dose estimates on the basis of excretion analyses.

P/398. Thyroid I^{131} from air-borne fission products. L. Van Middlesworth (U.S.A.).

I^{131} in thyroids of grazing animals has been followed continuously for the past 4 years. More than 9000 animals have been studied. Specimens are received regularly from Tennessee, Washington State, Australia, Japan, Germany and England. Large releases of air-borne fission products invariably have resulted in exponential increases of thyroid I^{131}. The following information is available from these data:

(1) After release of large amounts of I^{131}, the average distribution of radioactivity in thyroids of grazing animals is greatly influenced by the time between release and initial intake of the I^{131}.

(2) There are at least two major routes of intake of radioactive iodine in the animal; (a) respiratory (10% or less) and (b) ingestion (90% or more).

(3) I^{131} in human thyroids averaged 0.005 times the average concentration in cattle from the same area.

(4) Sheep thyroids usually contain 4 to 5 times greater I^{131} concentration than cattle.

(5) Effective half-life of I^{131} can be determined in various animal populations by this method. In cattle the I^{131} effective half-life has been 7 to 8 days, and in sheep 4 to 6 days.

(6) Radiation doses of I^{131} to the animal populations have been determined by this method.

(7) Thyroids of bovine fetuses (50 specimens) contain less chemical iodine but 1.5 to 2 times more I^{131} concentration than thyroids of pregnant cows; even 8 weeks after equilibration with air-borne I^{131}.

(8) This procedure has yielded useful physiological, ecological and radiological information. This method is probably the simplest and most rapid index to the biological concentration of air-borne fission products.

P/898. New developments in radiation protection and recovery. Alexander Hollaender, C. C. Congdon, D. G. Doherty, Raymond Shapira and A. C. Upton (U.S.A.).

Since the first Geneva Conference, several important developments have taken place in the field of radiation protection and recovery. This report will analyze these advances and try to evaluate their practical significance.

It has been found that it is possible to encourage healing of chromosomes that are broken by radiation by supplying an energy source after irradiation. This has been shown on bean roots and apparently it also applies to *Drosophila* flies. It appears that, at least in the limited number of organisms studied to date, although oxygen during irradiation will increase the radiation damage, oxygen supplied after irradiation will encourage the repair of the damage.

Among the many classes of chemicals that have been tested as protective agents against ionizing radiation damage, the most effective in mammals continues to be the mercaptoalkylguanidines, some of which may be administered orally as well as parenterally. The $LD_{50/30\ days}$ of X-rays for mice has been slightly more than doubled (from 700 r untreated to 1500 r treated). A maximum compound dose offers protection for up to six hours after administration at the 900 r level. Extensive chemical modifications have brought out interesting points about the relationship between protective activity and molecular structure; for example, the nitrogen and sulfur atoms cannot be separated by more than three methyl groups without loss of activity. A new group of compounds has been found which will protect at very low dose levels. On the basis of studies with radioisotopes, these compounds are selectively distributed within the animal and within the cell at sensitive biological sites and apparently function there as protective agents by acting as radical traps. They are also protective against radiomimetic agents such as mustard.

A field of radiation protection which has developed very rapidly during the last three years is the transplantation of bone marrow. With X-rays or γ-rays, bone marrow therapy allows nearly doubling of the $LD_{50/30\ days}$ in mice. It works less well with neutrons. Apparently the injected bone marrow settles out in the destroyed blood-forming tissues of the host and repopulates the marrow spaces. New blood elements are quickly formed, preventing the appearance of infection, hemorrhage and anemia. The speed of recovery is determined by the quantity of bone marrow injected. Bone marrow may be preserved by special freezing techniques.

Combined treatment with bone marrow and chemicals affords significantly better protection than treatment with either type of agent alone.

When the bone marrow is immunogenetically histo-incompatible with the recipient, a secondary disease (the delayed foreign bone marrow reaction) often appears

during the recovery from radiation injury. Many die from this secondary disease. A similar, more rapid, reaction occurs when foreign lymphoid or spleen cells are implanted. In case of lasting recovery with the donor bone marrow, other tissues from the donor can also be transplanted such as skin and tumor.

The production of long-delayed effects such as graying of the hair, kidney damage, tumor formation and shortening of the life span, is inhibited by chemical protection. Bone marrow does not protect against these long-delayed effects, except in leukemia, but it permits the animal to survive a reasonable length of time.

There are many complicated problems of immunology connected with bone marrow transplantation which are now being investigated extensively in animals, as well as in human patients. Chemical protection is also being investigated in man. Although the preliminary results are encouraging, it will take a number of years before we can adequately evaluate the clinical usefulness of these procedures.

P/887. Estimation of the turnover equation of strontium-90 for human bones. PATRICIA DURBIN and HARDIN JONES (U.S.A.).

Evaluation of radiation exposure from Sr^{90} needs to be based upon knowledge of its residence time in bone. Complete information on bone turnover in human beings is lacking, and to evaluate the probable equation of Sr^{90} turnover in man the turnover values for Ca^{45} and Sr^{90} have been measured in adult rats and monkeys (rhesus). The turnover equations for Ca^{45} and Sr^{90} in the rat and monkey are as follows:

Percent Dose Per Organ in Skeletal Parts

Ca^{45}—*Rat*

Vertebrae	$0.4\% \exp(-0.693d/3) + 3\% \exp(-0.693d/180)$
Mandible	$3.5\% \exp(-0.693d/330)$
Long bones	$0.9\% \exp(-0.693d/1) + 1.8\% \exp(-0.693d/12) + 4.5\% \exp(-0.693d/450)$
Whole skeleton	$4.4\% \exp(-0.693d/2) + 6\% \exp(-0.693d/6) + 67\% \exp(-0.693d/570)$

Sr^{90}—*Rat*

Vertebrae	$0.8\% \exp(-0.693d/13) + 1.83\% \exp(-0.693d/180)$
Mandible	$0.4\% \exp(-0.693d/16) + 2.4\% \exp(-0.693d/330)$
Long bones	$1.3\% \exp(-0.693d/3) + 0.9\% \exp(-0.693d/15) + 3.0\% \exp(-0.693d/450)$
Whole skeleton	$7\% \exp(-0.693d/2) 12\% \exp(-0.693d/12) + 43\% \exp(-0.693d/570)$

Sr^{90}—*Monkey*

Whole skeleton	$13\% \exp(-0.693d/3) + 11\% \exp(-0.693d/6) + 10\% \exp(-0.693d/50) + 38\% \exp(-0.693b/970)$

Discussion

Strikingly similar skeletal elimination rates of Sr^{90} and Ca^{45} were observed in the rat, especially in comparisons based upon whole-skeleton values. In the rat there appear to be different characteristics of bone-mineral turnover in the flat and in the long bones which can be reasonably related to the differential growth properties of the individual bones. It is apparent, however, that a gross similarity exists between the separate bones and the entire skeleton.

A remarkable similarity was found when the Sr^{90}-turnover equations for the rat and the monkey were compared, and this similarity suggests that the turnover equation for the monkey may be a reasonable estimation of turnover of Sr^{90} in human bones. One may then estimate the fractions of adult human bone that equilibrate with Sr^{90} and the rate at which such equilibration occurs.

These are:

0.06% of bone turnover with a half time of 3 days
0.2% of bone turnover with a half time of 6 days
1.3% of bone turnover with a half time of 50 days
98% of bone turnover with a half time of 970 days

The average cumulative radiation in bone following acute exposure to Sr^{90} is largely determined in that portion of the skeleton in which turnover rate is slowest; in this case 98% of the bone accumulates 38% of the administered Sr^{90} and subsequently eliminates it with a half time of 970 days. It is noted that for man an average Sr^{90} turnover in bone of about 1000 days would give a reasonable agreement with the Sr^{90} content of human bone as reported by Libby and others over the period from 1953 to 1957.

A human being exposed initially to a bone burden of 1 mμC of Sr^{90} (assuming uniform distribution in a bone mass of 700 g) would have the following retention of Sr^{90} and radiation exposure (if he were not subsequently re-exposed to Sr^{90}).

Time (yr)	Sr^{90} (mμC)	Dose (mr/yr)	Accumulated exposure (mr)
0 (initial exposure)	1	2.8	0
2.7	0.5	2.0	5.4
5.4	0.25	1.0	8.1
8.1	0.13	0.5	9.5
∞	0	0	10.8

P/401. Strontium-90 and cesium-137 uptake by vegetation under natural conditions. S. I. AUERBACH and D. A. CROSSLEY, JR. (U.S.A.).

The strontium-90 and cesium-137 movement from soil to man is one of the complex problems arising from reactor operations. Most of the information on the plant to soil relationships of these two fission products has been obtained from laboratory experiments utilizing (a) nutrient solutions, (b) prepared soils and (c) soils contaminated by weapons fallout. Little work has been done on these relationships with plants grown under field conditions.

Long-term studies on permanent areas contaminated with these radionuclides have not been reported.

Oak Ridge National Laboratory has an area which is uniquely suited for long-term studies under field conditions. This area was formerly a 40-acre radioactive waste impoundment (White Oak Lake) which for 12 years received a continuous input of low-level radioactive wastes. Two years ago the basin was drained, leaving about 35 acres of contaminated terrain. The resulting soil in this area can be characterized as a relatively unconsolidated alluvial sediment, heterogeneous in composition, calcareous, somewhat saline and alkaline in reaction. Concentrations of strontium-90 and cesium-137 in the first six inches range from 0.02 to 0.29 and from 0.5 to 2.0 microcuries per 100 grams of soil, respectively. Significant concentrations of cobalt-60, ruthenium-106, cerium-144 and the trivalent rare earths are present also. The $\mu\mu$c Sr-90 to 1 gram Ca and the $\mu\mu$c Cs-137 to 1 gram K ratios in the leaves of one of the native plants (*Polygonum lapathifolium L.*) on the lake bed ranged from 5.64×10^4 to 1.91×10^5 and from 7.73×10^4 to 1.67×10^5, respectively.

As a part of the investigations of uptake by vegetation, four common varieties of corn (*Zea mays*), namely, Hickory Cane, Aristogold Hybrid, Golden Bantam and Country Gentleman, were planted in a part of the lake bed laid out in a latin square. The general chemical properties of the soil in this plot were typical of most of the lake bed. The pH ranged from 6.86 to 7.50 and the salt pH from 6·66 to 7.21. Ammonium acetate extractions of calcium varied from 14.8 to 18.8, magnesium from 1.13 to 1.47, potassium from 0.15 to 0.17, and sodium from 0.14 to 0.16 M.E. per 100 grams of soil. Phosphorus ranged from 0.52 to 1.52 ppm. Ratios of strontium-90 to calcium were from 9.81×10^4 to 1.08×10^5; ratios of the $\mu\mu$c Cs-137 to 1 gram of potassium were from 4.67×10^6 to 8.20×10^6.

Chemical and radiochemical (particularly strontium-90) analyses for the corn plants and native vegetation have not yet been completed nor have the data as yet been given a thorough statistical analysis. However, the first results indicate that the concentrations of cesium-137 may differ between plant organs. The cesium-137 to potassium ratios in the leaves, husks, cobs and grain have the following ranges: Leaves, 2.93×10^4 to 5.87×10^4; husks, 2.18×10^4 to 4.18×10^4; cobs, 1.76×10^4 to 2.0×10^4; grain, 1.54×10^4 to 1.84×10^4. These data indicate a considerable discrimination against cesium-137 on the basis of 1 N ammonium acetate soil extractions of these cations. The strontium-90–calcium ratios obtained in this calcareous soil varied with the mode of extraction of these cations. The range of variation compares favourably with that reported by other workers.

P/430. Exposure criteria for estimating the consequences of a catastrophe in a nuclear plant. J. B. H. Kuper and F. P. Cowan (U.S.A.).

As a part of a study to assess the possible consequences of a catastrophe in a large nuclear power plant, estimates were made of the injuries to be expected as a result of exposure to a fission product cloud, and of the effects of ground contamination. A hypothetical 500 Mw (thermal) reactor was assumed to have been operating for six months. Two types of accident were postulated, one in which a large fraction of the total fission product inventory was released, and the other involving the escape of the volatile fission products only. Whole body exposures to beta and gamma rays, lung exposure to beta rays and the exposures of body organs to deposited isotopes as a result of direct exposure to the fission product cloud were estimated. An attempt was made to reduce these to equivalent values of whole body exposure, to combine them and thereby to obtain an overall index of biological damage. It was concluded that for the full fission product release an exposure to more than 400 curie-seconds per cubic meter (as measured at 24 hours after the accident) would likely be fatal, between 90 and 400 c-sec/m^3 illness would be expected, between 10 and 90 injury would be unlikely but there might well be expenses involved in establishing absence of damage and below 10 c-sec/m^3 no illness or expense would be anticipated. For the volatile fission product release the corresponding numbers are 350, 80 to 350, 10 to 80 and 10 c-sec/m^3.

Estimates of the effects of contamination on land use are necessarily rather crude, and actions taken would undoubtedly depend on the size and character of the area affected. Again various components of external and internal exposure were considered. They were combined and criteria for evacuation and limitations on the use of land were proposed. It was concluded that for the full fission product release areas contaminated with more than 10^{-2} curies (measured at 24 hours) per square meter would have to be evacuated, that between 10^{-3} and 10^{-2} c/m^2 severe restrictions on land use and outdoor work would be required, between 10^{-4} and 10^{-3} some crops might have to be destroyed and temporary restrictions on agriculture would be expected and that below 10^{-4} no restrictions would be necessary. For the volatile fission product release these figures would all be a factor of 10 greater.

P/548. Dosage du strontium-90 en presence de strontium-89 et de strontium-91. R. A. Wolschrijn et J. Kooi (Pays-Bas).

Dans certains cas, il est important de déterminer la teneur en strontium-90 dans des échantillons renfermant les trois isotopes 89, 90 et 91 du strontium.

Dans des produits de fission récemment formés, l'activité d'un échantillon de strontium isolé est due principalement au strontium-91 ($t_{\frac{1}{2}} = 9{,}7$ h). D'autre part, le strontium-89 ($t_{\frac{1}{2}} = 51$ d) présente une activité beaucoup plus grande que le strontium-90 ($t_{\frac{1}{2}} = 28$ ans).

On peut doser le strontium-91 en isolant l'yttrium immédiatement après avoir obtenu l'échantillon de strontium composé et en mesurant les rayons gamma de 0,55 MeV de l'yttrium-91 m.

Pour obtenir les quantités relatives de strontium-89 et de strontium-90, il faut isoler à nouveau l'yttrium après un laps de temps plus long et effecteur une correction sur l'yttrium-91 m pour tenir compte de l'yttrium-91.

La possibilité d'appliquer cette méthode à divers cas

sera discutée ainsi que le degré de précision que l'on peut en attendre. Des résultats expérimentaux seront exposés.

P/549. A simple procedure for the determination of Sr^{89} and Sr^{90} in urine and seawater. J. Kooi (Netherlands).

Recently a method for the determination of radioactive strontium in water samples has been developed. Herein, concentration is achieved by direct precipitation and total strontium content is obtained in about 5 hours, Sr^{90} in about 24 hours. The method does not require any special reagents or apparatus and is very well suited for routine application. Finally it is very selective and sensitive to less than one tenth of the M.P.C. for Sr^{90}. Radiobarium may be determined almost without any additional effort.

As some of the advantages mentioned are of still more importance in the routine analysis of samples of urine and seawater, the method has been examined for use in these cases. Experimental data will be presented to prove its applicability. The necessary changes in the procedure will be given in full detail.

P/739. The questions of health hazards from the inhalation of insoluble uranium and thorium oxides. Harold C. Hodge and Robert G. Thomas (U.S.A.).

The insoluble compounds of uranium and thorium (particularly the oxides) are important in the development of atomic energy. The questions of health hazards from exposures to dusts of these insoluble compounds are strikingly similar in many but not all respects. Among the similarities may be listed the following facts: (1) The insoluble compounds present no chemical hazard. Both uranium and thorium dioxides, for example, are remarkably inert physiologically. (2) No radiation injuries have so far been described in the lungs of experimental animals inhaling dust concentrations many times the recommended MAC. The lungs of a few dogs studied seven years after excessive inhalation exposures to ThO_2 gave negative histological findings although high concentrations of thorium were present. (3) The MACs for insoluble uranium and for insoluble thorium dusts are identical, specifically 3×10^{-11} μc/ml. Calculated on a radiation basis, a lower MAC is appropriate for thorium.

P/744. The calorimetric determination of local absorbed dose. P. Milvy, S. Genna, N. Barr and J. S. Laughlin (U.S.A.).

The development of a calorimetric technique for the determination of the intensity of radiation has been previously reported. This method has been further developed and applied to the measurement of locally absorbed dose. Its use for the determination of the absorbed dose produced in tissue-equivalent material by cobalt-60 gamma rays is reported here.

The entire irradiated medium under study is contained in the calorimeter. At the point where the absorbed dose is desired, a small segment of the medium is thermally isolated by means of vacuum gaps which are sufficiently small to prevent any perturbation of the gamma-ray absorption or of the secondary electron equilibrium. Quasi-adiabatic thermal control is maintained continuously. Imbedded in the absorbing segment is a thermistor which responds to temperature changes and whose output is electronically amplified and recorded. The temperature change has been calibrated in terms of the absorbed dose by means of the dissipation of a known amount of electrical energy. The measurements are reproducible with a standard deviation of less than $\pm$ one per cent.

The absorbing medium employed in the present investigation is fabricated of conducting plastic, and has as its constituents an equal number of carbon and hydrogen atoms. For cobalt-60 gamma-rays it is closely tissue-equivalent.

The major correction is for the energy absorbed in the crystal-lattice which does not appear as thermal energy, but as potential energy of chemical configuration. The results of Lawton *et al.* on the radiolysis of polyethylene were used to calculate a 2.36 per cent correction for the absorbed energy which is not thermally degraded.

In addition to the calorimetric determination of absorbed dose in tissue equivalent plastic, an alternative technique for the determination of this quantity, the Fricke Ferrous Sulphate Dosimeter, has been utilized. A G value of 15.5 molecules/100 eV was employed. A small teflon container was constructed, with the same dimensions as the plastic absorbing volume of the calorimeter. This was inserted in a polystyrene absorbing cylinder with dimensions equivalent to those of the calorimeter. The locally absorbed dose determined chemically and calorimetrically agreed to within 1.5 per cent.

The local absorbed dose calorimeter is physically direct and experimentally precise. It permits the measurement of the locally absorbed energy with a technique which avoids some of the problems encountered in the use of cavity ionization measurements for this purpose.

P/753. Conducting plastics equivalent to tissue, air and polystyrene. Francis R. Shonka, John E. Rose and Gioacchino Failla (U.S.A.).

A series of conducting plastic resins has been developed which make possible the direct measurement of absorbed dose in rads for photons and neutrons in muscle bone, and polystyrene; rads for photons in muscle (insensitive to neutrons); and roentgens.

The components chosen for tissue, air and polystyrene equivalent materials should result in a conducting blend with the best possible mechanical properties and having the same absorption for photons and neutrons as the material simulated. This means that for equal numbers of electrons in simulated and equivalent materials, the number of hydrogen and nitrogen atoms in the equivalent must be equal to those contained in the simulated material or to zero, depending on whether the equivalent material is to be sensitive or insensitive to neutrons; and the sum of the photoelectric absorption crosssection for soft X-rays must be equal for both.

Commercial plastics are blended with suitable additives producing compounds which achieve the desired atomic composition and radiation response, and which can be

readily produced for injection and compression moulding. So that the same photoelectric absorption coefficients for low energy X-rays will result, the oxygen or nitrogen deficiencies caused by high carbon content (including free carbon for electrical conductivity) must be balanced. This is done by substituting elements of higher atomic number; specifically, fluorine, silicon or calcium.

Since commercial equipment for intensive mixing in small batches is unavailable, a mixing system consisting of concentric screws and cylinders with but one rotating element has been designed to achieve the necessary high degree of turbulence and shear. Mills of up to 375 gram capacity are operating successfully. The additives used in compounding the conducting plastic resins are uniformly dispersed during mixing and are of a particle size less than one tenth of the range of the photoelectrons produced in them by soft X-rays.

The best compositions achieved to date are:

1. a. Tissue (muscle) equivalent for all radiations
$(C_{6.34}H_{10.2}O_{.32}N_{.25}F_{.112}Si_{.033}Ca_{.056})$
   b. Tissue (bone) equivalent for all radiations
$(C_{4.50}H_{6.4}O_{.19}N_{.19}F_{.89}Ca_{.446})$
   c. Polystyrene equivalent for all radiations (CH)
2. Tissue (muscle) equivalent for photons only
$(C_{5.34}F_{2.559})$
3. Air equivalent for photons
$(C_{138.5}H_{171.4}O_{29.6}N_{15.6}Si_7)$

Based on Grodstein [Nat. Bur. Standards 583 (1957)], the percentage error for all compositions is less than 5% for photon energies 10 keV and up, with the exception of 1a, where the error is slightly higher.

These thermoplastics absorb neutrons and photons over a wide range of energies in nearly the same way as muscle, bone, air and polystyrene, and therefore have important applications. One of these applications is for ionization chambers. Here, when the gas volume is large, it is necessary to use a gas mixture equivalent to the wall. Furthermore, if chambers are to read exposure dose, compensation for attenuation and degradation of the primary beam in the wall must be considered. Experiments are presently in progress to study whether the necessity for having equivalence of wall and gas can be eliminated by suitable choice of the geometry of the electrodes and adjustment in the composition of the wall.

P/756. Detection of plutonium in wounds. W. C. Roesch and J. W. Baum (U.S.A.).

A problem of long standing at plants handling plutonium is its detection in wounds from which the alpha particles cannot penetrate. About 4% of the disintegrations of plutonium are followed by emission of X-rays of about 17 KeV. Thin RaI crystals are very efficient detectors of these rays (0.25 mm gives 95%) and have low backgrounds. Three cases of plutonium in wounds have been studied. Depositions of 0.0014 to 0.02 microcuries were found. After excision or strenuous decontamination, it was shown that less than 0.00004 microcuries was left.

P/757. A summary of data that was used in the revision of the internal dose recommendations of the international commission on radiological protection. K. Z. Morgan, I. H. Tipton and M. J. Cook (U.S.A.).

The report of the Internal Dose Committee of the International Commission on Radiological Protection has been revised and released for publication (expected date November 1958). The basic principles and fundamental philosophy of radiation protection are unchanged in this new publication, but a number of revisions are made. The most important of these is the reduction of maximum permissible concentration, MPC, of radionuclides in air, water and food so that the average RBE dose rate to the total body of the occupational worker will be one-third of that formerly permitted. This reduction is intended to reduce the probability of chronic damage, e.g., leukemia and premature aging.

The MPC values for bone-seeking radionuclides continue to be based on an accumulation in the bone that is estimated to deliver an RBE dose rate equal to that received from 0.1 μg of Ra^{226} and its daughter products. These values were reduced by 32% because recent data indicate that only 30% (rather than 55%) of the daughter products of Ra^{226} is retained in the body. The new MPC values correspond to an RBE dose rate of 0.56 rem per week to the bone, 0.6 rem per week to the thyroid or skin, 0.1 rem per week to the total body or gonads and 0.3 rem per week to all other organs of the body.

A major objective in the revision of the ICRP Handbook has been to obtain reliable values for the distribution of radioelements in the human body following many years of exposure. One of the best sources of such data is the normal distribution of stable isotopes of these elements in the human body. For the past six years a study of human tissue has been conducted by the Oak Ridge National Laboratory in cooperation with The University of Tennessee. This study includes a spectrographic analysis for 37 elements in 35 body organs, from over 300 autopsies that were supplied from 15 countries. For elements such as Ca, Fe, Zn, Mn, etc., which are known to have a role in the body economy, the geographical variation is not large—reflecting the state of nutrition of the population. For other elements such as Sr, Cd, Pb, Al and Cr which may compete with necessary elements in their normal roles the geographical variation is striking. Some elements such as Cu and Zn in liver, Al in lung and Cd in kidney, show marked variation with age of subject. Such data are useful not only in calculating the MPC values for the standard man but may serve also as a guide in making refinements to the MPC values when applying them to individuals of a given age, weight, eating habits, etc.

MPC values are given for 86 radionuclides in the 1955 ICRP Handbook and for 230 radionuclides in the 1958 edition. Also, MPC values are given for insoluble as well as soluble radioactive materials and for several critical body organs resulting in a seven-fold increase in the number of MPC values. In a few cases (where sufficient data are available) the MPC values are based on a power function rather than an exponential function of the time of exposure.

P/759. A sensitive analytical method for the determination of very low level plutonium in humans. L. C. SCHWENDIMAN and J. W. HEALY (U.S.A.).

The expanding nuclear power economy will result in ever increasing potential for internal deposition of plutonium in workers. This will occur as a result of fuel element reprocessing in ever increasing quantities with consequently more and more individuals potentially exposed to plutonium. An important element of protection is the early detection of plutonium deposition as indicated by the appearance of small quantities of plutonium in urine. A method was developed for determining extremely low levels of plutonium in urine. This method involves sample evaporation, muffling, dissolution of salts, coprecipitation with lanthanum fluoride and chelation with thenoyltrifluoroacetone to isolate the plutonium. An electrodeposition method subsequently separates and deposits the plutonium uniformly on a small known area of a disc. Alpha track film is used to record the emission of alpha particles over a period long enough to give requisite sensitivity. After emulsion development, the tracks are counted under a microscope. A detection limit on the order of 0.03 d/m/sample (2×10^{-13} g plutonium) is achieved. This detection limit permits recognition of exposure at a level of 0.005 of the maximum permissible body burden, thus insuring timely detection of incipient deposition.

P/762. Low intensity spectrometry of the gamma radiation emitted by human beings. C. E. MILLER and L. D. MARINELLI (U.S.A.).

As a consequence of the widespread use of radioelements in medical and industrial research, and of the prospective reliance on nuclear fission for the generation of useful power, the probability of contaminating the human body with radioactive materials is bound to increase. Since this contamination must be kept within limits consistent with the safety and welfare of the individual, it is obvious that procedures capable of measuring radioactivity in the human body below these levels will be a substantial help in many fields.

In many instances, this situation can be aided materially by means of low level gamma-ray measuring techniques, which permit the evaluation of very low specific activities of many radioelements in specimens of relatively large bulk. In our laboratory, efforts have been concentrated on the development of gamma-ray spectrometry at low levels of intensity in order to attain both the sensitivity and selectivity necessary to various tasks.

To eliminate the interference of the natural radioactivity in the immediate surroundings, measurements are made within a $2 \times 2 \times 2$ meter enclosure with walls of laminated sheets of steel, 20 cm in thickness. NaI(Tl) crystals are used for gamma ray detection because of their superior spectrometric properties. Whole body levels are measured by an $8'' \times 4''$ thick crystal, placing the subject on an arc of a circle or the standard reclining chair. Collimated crystals of various sizes are used to scan the individual and determine the distribution of activity. Under favorable circumstances localized activities as low as 2×10^{-10} c have been measured with reasonable accuracy.

It is impossible to adequately cover within the limits of an abstract both the uses to which this instrument has been put or discuss its possibilities in medical investigation. Some illustrative examples, however, are in order. As to the field of industrial hygiene and medicine, it has been possible to study contamination in living humans that has accrued as a result of accidents in the laboratory; valuable data have been obtained on transit time of radioactivity in the lung following inhalation intakes of $RaSO_4$, Pa^{233} and Te^{129}. Contamination due to punctures of contaminants of Am^{241} and Th^{227} have also been followed for a long time and yielded, in the latter case, information on equilibrium of the radioactive chain *in vivo*.

It is obvious that these techniques will allow systematic studies of metabolism by multiple tracer techniques and that they are particularly well adapted to long term investigations using fairly long lived isotopes, since the radioactive burdens required are essentially similar to those encountered at natural levels; a case in point has been the study of Sr^{85} retention (65 days half life) for a period of 18 months following an initial ingestion of only 50 μc.

Another merit of this technique in the field of clinical medicine is the accurate determination of total body potassium by the radioactivity of its isotope K^{40}, devoid of the interfering activity of Cs^{137} which has been shown to vary with both the geographical location and the dietary habits of the individual under investigation.

P/763. Dosimetry of γ-ray and neutron fluxes in CP-5. EDWIN J. HART and P. D. WALSH (U.S.A.).

Water decomposition studies in the Argonne heavy water research reactor CP-5 illustrate the principles of γ-ray and neutron pile dosimetry. Sixty students of the International School of Nuclear Engineering obtained the data reported in this paper as a part of a special experimental course.

Pure water, irradiated by γ-rays and neutrons in the core of CP-5, is stable. In contrast, water contaminated by free radical scavengers or by nuclear poisons such as boron, decomposes into hydrogen and oxygen. These gases are liberated from the irradiated water in the ratio of 2 to 1 for steady state decomposition. The rate of gas evolution provides a measure of the neutron and γ-ray fluxes.

Solutions of potassium iodide, an efficient free radical scavenger, are used to monitor the γ-ray flux. And solutions of boric acid plus potassium iodide are used to find the sum of the neutron and γ-ray fluxes. The difference between the gas evolution rates for these two solutions yield data from which the thermal neutron flux may be calculated.

The γ-ray and neutron flux patterns in the core and reflector regions of CP-5 operating at 1 megawatt will be discussed. The neutron flux depends on vertical position in the reactor core, varying from 1.7×10^{13} $n/cm^2/sec$ in the center of a vertical thimble to 0.67×10^{13} $n/cm^2/sec$ 18 inches higher. γ-ray fluxes in the core range downward from 6×10^6 rad/min.

Iodide ion concentration has a marked effect on the rate of gas evolution. At a γ-ray dosage rate of 6×10^6 rad/min, at least $3 \times 10^{-4}M$ potassium iodide is needed to produce the maximum rate of water decomposition. Below this concentration, partial recombination of hydrogen and oxygen occurs.

Boric acid, in concentrations from 0.025 to 0.12 M, serves as a satisfactory neutron absorber. The lower concentration in this range is better for the fluxes in the core, the higher concentrations are better for the reflector fluxes.

Approximately 60 ml of the iodide and boric acid solutions are irradiated in silica vessels loaded in the reactor in thin-walled aluminum cans. Aluminum capillary tubing ($\frac{1}{4}''$ o.d., $\frac{1}{8}''$ i.d.) connects the silica cell with an external gas measuring apparatus. The rate of gas evolution is measured at constant pressure. Usually 15 to 30 minutes are required for a steady decomposition rate.

P/765. Treatment of plutonium inhalation case studies. W. D. Norwood, M.D., P. A. Fuqua, M.D., R. H. Wilson and J. W. Healy (U.S.A.).

With the increasing use of atomic energy in industry, medicine and defense, the problem of diagnosis and treatment of individuals, who have inhaled significant quantities of radioactive materials, becomes more urgent. Little information is available on the treatment of plutonium poisoning in humans. This paper attempts to contribute to the meager knowledge in this field by describing methods used and results obtained in treating two individuals, who accidentally inhaled a plutonium compound during the course of work at the Hanford Atomic Products Operation, Richland, Washington, U.S.A. Because of very effective control of contamination, this was the first significant overexposure to radiation in 13 years of operation.

In interpretation of the urinary excretion, the initial lung burden was estimated by a procedure which accounts for the rate of transfer to the blood stream. This permitted an estimation of the lung exposure to alpha radiation so that the possibility of lung damage could be estimated.

In the first case, treatment, following decontamination, was directed at increasing excretion via the urine and was continued intermittently over a period of seven months. Effectiveness of treatment was evaluated by daily analysis of 24-hour urine specimens for plutonium. Fecal elimination was also carefully followed and sputum specimens were occasionally checked for plutonium.

The principal method of treatment was by intravenous calcium–disodium ethylene diamine-tetra-acetate (CaEDTA), and this consistently increased the urinary output of plutonium by a factor of 10. Oral CaEDTA, both plain and with enteric coating, was found to be ineffective. Zirconium Citrate was occasionally used and found to be effective. Careful observation, and tests failed to reveal any adverse effects of this extended treatment.

The early fecal measurements indicated that about 0.34 microcuries were deposited in the upper respiratory tract, and eliminated by ciliary action, and mucous to the throat. Subsequent urinary excretion values over a period of 220 days indicated an additional deposit of about 0.36 μc which remained in the lung, and was transfered to other portions of the body with a half life of 30–40 days. In spite of extensive treatment, only about 0.04 μc were eliminated in the urine.

The internal deposit sustained by the second employee is estimated to be about 0.03 microcuries of plutonium. Treatment and results were similar to those obtained in the first case described.

The paper concludes that, (1) treatment of individuals who have inhaled plutonium compounds is apparently even more unsatisfactory than usual if the compound is poorly soluble, and slowly absorbed from the lungs; and hazard to lung is an added problem, (2) the study indicates the urgency for continued research to find more effective methods of treatment and (3) the most exhaustive control methods must be continued to prevent plutonium deposition.

P/911. The influence of strontium-90 upon life span and neoplasms of mice. Miriam P. Finkel, Birute O. Biskis and Gertrude M. Scribner (U.S.A.).

Predictions of the potential hazard to man of world-wide contamination with strontium-90 must be based largely upon the results of animal experimentation. Among the most useful criteria of radiation damage to the mammalian organism as a whole are decrease in life span and increase in the incidence of certain tumors. However, these effects can be studied and evaluated adequately only when large numbers of animals are observed for the duration of their natural lives. The mouse is admirably suited to this type of experimentation.

Although the human problem is primarily one of deposition within the body of material that has been ingested or inhaled, extrapolations from mouse to man are simplified if the radiostrontium is given by intravenous injection because of species differences in absorption. At low levels the effective dose is probably that which becomes incorporated in the skeleton rather than that which passes through the gastrointestinal tract or lungs. Differences in uniformity of distribution, depending upon whether the period of exposure is relatively short or long, have been studied by administering the total dose in a single injection or in multiple, fractionated injections. The twelve dosages administered to 1320 CF No. 1 female mice ranged from acutely lethal levels to one that resulted in a body burden corresponding to approximately 10 μc/man, or to 10 times the present maximum permissible level for occupational exposure. The control population consisted of 240 mice.

Neither the average life expectancy nor the time to 50% mortality plotted against the dose resulted in curves that could be described adequately by a simple linear function or by any of a number of mathematical transformations. At the three lowest levels the differences in survival between the experimental and the control populations were within the range of expected variation. A statistically significant difference in life expectancy was not seen below a dose roughly equivalent to 350 μc/man.

The data on osteogenic sarcomas similarly failed to demonstrate a linear dose-response curve, and the number of such neoplasms among the 3 lowest dosage fell within the control range. Thus life span and malignant tumors of bone appeared to be equally sensitive indicators of radiation damage. Neoplastic change of the blood-forming tissues, however, proved to be a more sensitive criterion of injury. Although the incidence of reticular tumors was not markedly influenced by radiostrontium, they appeared earlier as the dose increased. The time required to reach a 20% incidence was significantly shorter among the animals that had received 8·9 μc/kg than it was among the control animals. This injected dose resulted in a body burden of approximately 1 μc/kg, which is roughly equivalent to 70 μc/man. Below this level the treated animals were indistinguishable from their controls.

Since the differences in life span and neoplasms of bone and reticular tissues among the lowest doses and the control population were not statistically significant, it appears either that pathological change in the total mammalian organism is too crude a measure of toxicity at very low levels or that such levels actually are without effect.

P/971. Dose concepts and dose measurements. H. FÄNRZ and W. HÜBNER (Federal Republic of Germany).

In the dosimetry three different concepts of doses are used: the "absorbed dose" with the unit "rad", the "exposure dose" with the unit "roentgen" (r) and the "RBE dose" (RBE—Relative Biological Effectivness) which shall be specified in "rem" (Recommendations of the International Commission on Radiological Units and Measurements, ICRU).

Only the absorbed dose is defined in a physically unobjectionable manner, i.e. by reduction to known physical quantities: $D = \Delta E/m = \Delta E/\rho \,.\, \Delta V$ where ΔE means the energy imparted by the radiation to the body element of the mass Δm (ΔV volume, ρ density). The unit of this dose in the international system of units of the meter convention is the Wattsecond/Kilogram. As a special unit it is defined: 1 rad $= 10^{-2}$Ws/kg $= 100$ ergs/g. As every dose originates by absorption of a radiation—the vacuum is not able to receive a dose—the name "energy dose" is suggested for the quantity D in order to distinguish it from other concepts of dose.

The energy dose is defined for all kinds of ionizing radiations. But for cases of practical interest generally it is not measurable directly: the place of interest in the body is mostly not accessable for measurements, the amounts of energy are too small for calorimetric measurements. D depends on the kind of the material absorbing the radiation, especially for soft X-rays.

In the practical dosimetry of X- and gamma-rays the measurement is carried out within a standard material, namely in air, the absorption coefficient of which is dependent on the quantum energy of these radiations approximately in the same way as that of soft tissue. Thereby the ion charge produced by the radiation is measured under fixed conditions.

The method of measurement based upon the ionization of air will remain, as it was, the basis of dosimetry, but till now an exact definition of the concept of dose based upon this measurement is missing.

By its own words the ICRU defined the "exposure dose" only in loose terms. In the definition of the unit "roentgen" the definition of a quantity and of a unit are mixed in such an incomplete way that the physical dimension of the quantity does not result clearly. This fact caused different interpretations in the international literature and a criticism of the definition of the "r".

It is suggested to define the quantity $J = \Delta Q/m$ as "ion dose", where ΔQ is the ion charge produced by the radiation within the air volume ΔV of the mass $\Delta m = \rho \,.\, \Delta V$. The unit of ion dose in the international system of units is the Coulomb/Kilogram; as a special unit of ion dose it is defined 1 r $= 2.58 \times 10^{-4}$ C/kg (corresponding to 1 electrostatic unit of quantity of electricity/0·001293 g). If an ion dose by X- or gamma-rays of a quantum energy less than 3 MeV is measured under usual standard conditions i.e. with electronic equilibrium in air it shall be called "standard ion dose".

The advantages of these definitions are the following:

1. Within the conventional region of X-rays all methods measurements as well as all numerical relations will be unchanged if using standard ion dose.

2. The ion dose is also applicable to X-rays in the high voltage range (e.g. measuring build up curve of a betatron radiation).

3. The definition is valid also for each other ionizing radiation (α-, β- and proton-radiation) except neutrons which require special investigations.

4. By the measured ion dose the energy dose can be computed in a known way.

For protection purposes it is suggested to define, instead of the RBE dose, as "equivalent dose" that X-ray energy dose D_x which produces the same biological effect as e.g. the neutron dose D_n to be considered. Then the equation $D_x = \eta_{ln} D_n$ is valid, η_{ln} being the dimensionless relative biological factor of effectiveness. D_x as an energy dose of X-rays is measured in "rad". Thus the not clearly defined "rem" will be unnecessary.

P/1008. Radiostrontium metabolism and decontamination in man: chelation in biology and medicine. DANIEL LASZLO and HERTA SPENCER (U.S.A.).

I. RADIOSTRONTIUM METABOLISM AND DECONTAMINATION IN MAN

In view of the world-wide concern of Sr^{90} toxicity, metabolic data (absorption, tissue distribution and excretion) and decontamination in man are of paramount importance. Such data have been obtained in 70 adults studied under controlled conditions on the metabolic research unit. *Absorption*: an average of 20% of the ingested dose is absorbed. Addition of calcium, of whole or skimmed milk to the diet did not change the radiostrontium absorption. Radioactivity can be promptly detected in *plasma* after ingestion of Sr^{85}, the level being highest at 4 hours, declining rapidly thereafter and low concentrations being present several weeks. The *extracellular* fluid concentration and bone *uptake* of Sr^{85} were calculated.

Urinary radiostrontium excretion occurs promptly after ingestion of Sr^{85}, the rate of excretion being highest at 8 hours and declining with time. A relationship between urinary radiostrontium excretion and bone metabolism was noted. Radiostrontium excretion in man can be predicted by empirical equations based on excretion and renal clearance of calcium. The renal clearance of radiostrontium is in average five times higher than that of calcium.

Radiostrontium and radiocalcium metabolism were simultaneously studied in 20 patients: the absorption, plasma levels, extracellular fluid and bone concentration and the renal and intestinal excretion of strontium differ significantly from that of calcium. Therefore, radiostrontium toxicity in man cannot be determined via calcium metabolism. Extensive *decontamination* studies were carried out. Various chelating agents were ineffective. Increase of Ca^{++} in plasma by administering calcium in conjunction with metabolic acidosis were effective, decreasing the body burden significantly shortly after exposure and two weeks later.

II. Chelation in biology and medicine

Chelation develops into a promising field of biology and medicine. Ethylenediamenetetraacetic acid (EDTA) is widely used for removal of radioactive metals, e.g., lead, plutonium and yttrium. When lanthanum-EDTA was injected in tracer amounts intravenously to man (molar ratio of metal to chelating agent 1 : 1), the metalchelate became dissociated and less than 10% of the metal was excreted.

This observation necessitated a systematic study of metalchelate metabolism *in vitro* and *in vivo*. A series of metals and of 12 chelating agents having stability constants ranging from log K 6 to 30 were studied. Using equilibrium dialysis and pressure filtration of plasma and of purified protein fractions *in vitro* and tissue distribution and excretion studies *in vivo* a strict relationship was found between metalchelate metabolism and stability constants and metalchelate metabolism and molar ratios. Tissue uptake was found to depend on metalchelate stability. Urinary metal excretion was studied in man employing 12 chelating agents: it was possible to vary the excretion from 5% to ~100%. High excretions were achieved by chelating agents having a stronger binding power than EDTA, e.g., the excretion of yttrium was ~50% following the administration of tracer doses of Y-EDTA while it was 97% after an equivalent dose of yttrium diethylenetriamine pentaacetic acid. 1 mathematical analysis of biological data was developed based on metalchelate stability and molar ratios. Employing these equations, predictions of biological distribution *in vitro* and *in vivo* are possible. Clinical application of this principle will be discussed.

P/1029. Radiostrontium–calcium relations in plants and animals. H. A. Kornberg, W. J. Clarke, J. F. Cline, F. P. Hungate, R. F. Palmer and R. C. Thompson (U.S.A.).

In measuring Sr^{90} contamination of materials as a hazard control procedure, data are usually expressed as SR^{90}/Ca ratios. This implies that strontium and calcium in being transferred, for example, from a plant to an animal behave either precisely the same or different by a constant number which represents the value of the Sr^{90}/Ca in the acceptor divided by the Sr^{90}/Ca ratio in the donor. Further, it is usually assumed that strontium and calcium compete with each other for deposition in plants and animals, the result of this being that the deposition of strontium in an acceptor from a donor containing strontium is predicted to be inversely proportional to the concentration of the calcium in the donor.

In assessing the hazard of Sr^{90} in soil by means of the Sr^{90}/Ca ratio, it is further assumed that these two substances are equally available for uptake by growing plants. By a consideration of several plausible rates with which Sr^{90} from fallout or waste disposal mixes with soil to become as available for uptake as the calcium already present in the soil, it can be shown by semiquantitative theoretical treatment that this ratio will lead to probable errors in hazard assessment.

In laboratory experiments the uptake in plants of Sr^{90} was measured as a function of calcium in the root environment. In general, the uptake of Sr^{90} was not found to decrease as a result of the calcium present.

When rats were intraperitoneally injected or fed in single doses mixtures of Sr^{90} and calcium, the deposition of Sr^{90} on the skeleton was not found to be affected by the presence of calcium in the administered dose. Groups of rats which were maintained on diets containing several levels of calcium along with Ca^{45} and Sr^{90} were found to deposit less Sr^{90} and Ca^{45} on their skeletons as the calcium in the diet was increased. However, the decrease of Sr^{90} deposition with increase in dietary calcium was less than that which would have occurred if the strontium had deposited in inverse proportion to the calcium in the diet. The data also showed that beyond 0.5% of dietary calcium there was no further decrease of Sr^{90} deposition with increase of calcium.

Similar observations were made with lambs that were fed Sr^{90} and calcium gluconate in milk. The gluconate ion was found to account for a significant amount of the decreased Sr^{90} deposition observed.

These experimental observations point to the need for measuring Sr^{90} contamination in absolute units and tend to support theoretical considerations of the effect of isotopic dilution. The movement and deposition of substances can occur by only four processes: flow, diffusion, adsorption and chemical reaction. Each of these processes may be separately considered to determine the effect of the addition of one isotope on the movement or deposition of another isotope. For the processes of flow and simple diffusion, isotopic dilution is without effect. For the processes of adsorption and chemical reaction, isotopic dilution will be effective only under the condition that some acceptor material upon which the element deposits approaches saturation. Even under this condition the effect of one isotope on another need not be a linear inverse relationship. Experimental data involving radioactive and non-radioactive isotopes other than radiostrontium and calcium which also support these hypotheses will be presented.

P/1354. Detoxication and removal of radioactive strontium in the rat. Yosoji Ito and Susumu Tsurufuji (Japan).

(1) Theoretical equations for estimating the effects of chelating agents for inhibiting bone deposition of radiostrontium are given.

(2) In short period experiments, sodium tricarballylate, citraconate, tartrate, lactate, citrate and propane-1,1,2,3-tetracarboxylate selected according to the above-mentioned equation markedly inhibited bone deposition of radiostrontium and kept it remaining in plasma.

(3) Urinary excretion of radiostrontium was increased and bone deposition was decreased by sodium citrate (excretion 6 hours after radiostrontium injection: 1.9 times the control, bone deposition: 0.85 times the control), tricarballylate (2.6, 0.77) and propane-1,1,2,3-tetracarboxylate (1.7, 0.85). No risks of hypocalcemia were observed.

(4) Above-mentioned theoretical equations indicate that sodium calcium citrate is one of the most promising known chelators which would not give any risk of hypocalcemia. Solution of this salt (20m Mole/kg) supersaturated with calcium citrate, prepared by the authors' method, was given to rats with simultaneous injection of radiostrontium. Six hours after the above treatment, 59.7% of the injected radiostrontium was removed from the animal bodies and control and sodium citrate groups (2m Mole/kg twice) had excreted only 18.7 and 30.5% respectively. About 50% of the citrate injected in the form of sodium or sodium calcium salts was excreted into the urine, i.e., 136 or 646 times of the control respectively. Plasma citrate level of sodium citrate group increased to 8.5 times of the control 30 minutes after the injection, and under this condition, the rate of bone deposition of radiostrontium should theoretically decrease to about 65% of the control.

(5) Although several experimental conditions markedly elevated urine citrate through biochemical responses, the radiostrontium excretion did not increase in parallel with urinary citrate when there was no significant increase in plasma citrate.

(6) No difference was observed between P-deficient normal Ca and high Ca diets.

(7) P-deficient diets increased excretion of injected radiostrontium in both adult and young rats.

(8) Percentage of excretion and retention of injected radiostrontium and carcass Ca were investigated by using five different kinds of diet: Normal control (*Sr Excretion and carcass *Sr: 18.1 and 75.5% of injected dose), low-Ca diet with NH_4Cl (14.9%, 82.3%), low-P diet (75.9%, 20.3%), low-P diet with NH_4Cl (75.1%, 19.8%), low Ca and low-P diet (60.0%, 38.4%). These results indicated: (i) P-deficient diet was one of the most efficient treatments to remove radiostrontium, (ii) NH_4Cl gave no effect, (iii) bone resorption was not necessarily an essential condition to remove radiostrontium, and there may be appropriate treatment for removing radiostrontium without bone resorption.

(9) Mildly P-deficient diet increased radiostrontium excretion and decreased body retention very effectively without bone disturbances.

(10) Statistically significant correlation was recognized between urinary Ca and radiostrontium excretion.

(11) Marked increase of both intestinal absorption and urinary excretion of Ca and increased probability of renal transit of blood radiostrontium were observed in rats fed on low-P diet. These were principal driving forces for removing radiostrontium.

(12) Radiostrontium excretion was markedly increased by a diet that was low in both P and Ca and contained $SrCO_3$. Effect of this diet was attributed not only to increase in renal transit but also increase in the rate of liberation of radiostrontium from the bone.

P/1859. Solid state dosimeters for radiation measurement. James H. Schulman (U.S.A.).

Exposure to ionizing radiation can produce changes in a number of the physical properties of insulating solids. Changes in optical properties—specifically in absorption and luminescence—are especially striking with some materials, and these effects have been extensively investigated for application to dosimetry. Resulting from these investigations are dosimetric devices characterized by great convenience in handling, long and perhaps unlimited shelf life, small volume and ruggedness. The dose range that can be measured by solid state detectors is literally enormous, extending from the order of milliroentgens to beyond 10^8 roentgens, depending upon the phenomenon employed.

All solid state dosimeters are secondary instruments and must be calibrated against some standard, such as an ion chamber. Because of the definition of the roentgen solid state dosimeters exhibit an energy-dependent response, particularly at low energies. This is especially serious with inorganic solids because of the comparatively high atomic numbers of their constituents. Selective shielding of these devices has been successfully employed in order to achieve a reasonable energy independence.

The general theory underlying inorganic solid detectors may be summarized as follows. When an electron is freed from an atom by irradiation of a solid, it may wander through the material until it is "trapped" by some constituent, such as a chemical impurity, or by some region of the solid containing a flaw or defect in the structure. The entity or "center" formed by combination of the electron with the trapping site has different optical properties from the normal solid, i.e., it may absorb light in a spectral region where the normal solid is transparent (coloration), it may emit fluorescent light under conditions where the normal solid would not do so (radiophotoluminescence, thermoluminescence or stimulable luminescence), or it may fail to fluoresce under conditions where the normal solid would emit light (degradation or quenching of luminescence). These radiation-induced changes are detected by optical techniques and constitute a measure of the dose.

The centers are responsible for coloration, radiophotoluminescence, or the degradation of luminescence are generally quite stable, permitting repeated measurement of the dose if desired. The centers giving rise to thermoluminescence or stimulated luminescence are necessarily

destroyed by the measurement procedure, so that the dose reading cannot generally be repeated.

Dosimeters based upon all the above-mentioned effects will be described, their operational advantages and limitations will be delineated and their applications will be discussed in personal monitoring, internal dosimetry and in the radiation-preservation of foods.

P/1987. Health physics aspects of handling U^{233} feed material. JAMES N. P. LAWRENCE (U.S.A.).

The alpha and gamma radiation problems, which have been encountered at Los Alamos Scientific Laboratory in the chemical purification, casting and machining of U^{233} feed material, are treated. The high specific activity of this material as compared to the other uranium metals encountered require the working of U^{233} feed in dry boxes. The permissible body burden (q) and $(MPC)_{air}$ of soluble forms which Dr. Morgan *et al.* have prepared for the 1958 Geneva Conference, are restated and the $(MPC)_{air}$ is converted to occupational exposure level of 136 d/m-M^3. The average air count in chemical processing line at LASL for 1956 and 1957 was only 4 d/m-M^3. An accident is discussed which resulted in a high air count of 1400 d/m-M^3.

The philosophy of the use of respirators as means to prevent intake of U^{233} into the body is discussed. The nebulous meaning of urinalyses for this material is described taking into account the rapid elimination immediately after uptake.

At LASL the significant level of U^{233} in the urine has been established as 50 d/m-1. Since 1956 the highest sample encountered has been 75 d/m-1, which was not maintained. The average for those persons working exclusively in the U^{233} chemical processing line was 13 d/m-1.

The gamma problem in handling U^{233} feed arises from the U^{232} daughters present. The first daughter of U^{232} is the third daughter of natural thorium. It would require, at equilibrium, about 3750 times the mass of natural thorium as U^{233} feed material, with only 20 ppm of U^{232}, for the same unshielded gamma emission.

The gamma problem is one of growth, which maximizes at about 10 years after the last chemical purification. Calculation data on the output in r/hr at 1 cm from a $\frac{1}{4}$ inch thick, 2 inch diameter disc of U^{233} feed (20 ppm U^{232}) indicates:

Age:	*1 day*	*4 weeks*	*32 weeks*	*10 years*
U^{233} plus daughters	0.0029	0.0032	0.011	0.15
U^{232} plus daughters	0.019	0.37	3.2	14.0

A disc $\frac{1}{4}$ inch thick, 2 inches in diameter was measured by extrapolation chamber at 28 days after casting. A surface dose of 6·6 rad/hr was found. At a year and 2 months later about 55 rad/hr was found.

For material $\frac{1}{4}$ inch thick or more about $\frac{1}{2}$ of the total gamma dose output is due to the 2.8 MeV gammas of thallium-208 (the seventh daughter of U^{232}). Actual measurements of material $1\frac{1}{4}$ years old indicated that $\frac{1}{4}$ inch of lead shielding reduces the output to only $\frac{1}{2}$ of the unshielded value. Additional $\frac{1}{4}$ inches of lead reduce it only by factor of 0.7 each.

Batches of 250 grams in about 2 liters of solution, which read from 0.1 r/hr to 3.0 r/hr, have been chemically processed in the present chemistry line.

During 1957 the total gamma exposure received by all nine persons performing the chemical purification was about 44 rem. In 1956 a total of 18 rem was recorded; however, about four times more quantity of material was processed in 1956 than in 1957. This indicates an increase of initial radiation rate in 1957. Some individual exposures are discussed comparing the years 1956 and 1957, pointing up the very few bi-weekly overexposures which have occurred.

The shielding presently used in the chem. line is mentioned, and the newly designed, heavily shielded dry box, which will be used for an ion exchange step in the chem processing, is discussed.

The metal fabrication is done in areas in which plutonium is handled, thus adding no additional restrictions to the previously imposed permissible air concentrations.

A typical exposure record for persons casting and machining U^{233} feed material covers five persons over a period of seven weeks, when about 15 kg of material was worked. They received an average of 0.7 rem each per week. During this time five weekly overexposures occurred.

In order to process this material on a routine basis more remote control handling, additional shielding in dry boxes, more personnel to spread the exposure and less delays from the initial chemical processing to the final machining would be necessary.

P/2058. De la possibilité d'enfouir les déchets radioactifs dans les fosses océaniques profondes. V. G. BOGOROV et E. M. KREPS (U.R.S.S.).

Le rejet des déchets radioactifs dans les fosses océaniques profondes repose sur l'hypothèse que les masses d'eau abyssales ne subissent pas de circulation verticale ou horizontale.

Les résultats d'observations portant sur 12 fosses océaniques, en particulier de la fosse de Tonga, ont montré qu'il existe dans toutes ces fosses une circulation, lente il est vrai, mais néanmoins mesurable non pas par siècles mais par dizaines d'années.

Cette conclusion repose sur une étude du relief de la fosse, de la distribution des températures, de la salinité, de la densité de l'eau, de la répartition de l'oxygène et des phosphates, ainsi que de la vie dans le fond de la fosse, des phénomènes biochimiques sur le fond et enfin de la circulation des eaux dans cette région de l'océan.

P/2078. Experiments designed to substantiate the maximum permissible doses of thermal neutrons. A. G. ISTOMINA, and I. B. KEIRIM-MARKUS (U.S.S.R.).

The adopted maximum permissible levels of thermal neutron fluxes have not been sufficiently approved experimentally. The theoretical calculation is based on approximate models. Therefore experiments have been carried out to model a gamma ray field of the capture of thermal neutrons and a proton field from a reaction on nitrogen. The experiments are made on a paraffin phantom of the human body. The gamma rays of neutron capture by

hydrogen are imitated by the gamma radiation of radioactive sodium-24; for this purpose soda is specially mixed in with the paraffin. The gamma field distribution is investigated by the photomethod. The proton field is identified with the thermal neutron field, which has been studied by the foil activation method.

It is established that in an isotropic flux of thermal neutrons the gamma ray dose is constant inside the body, somewhat increases when approaching the surface, and again decreases on the surface. The dose of protons monotonically decreases towards the centre of the body. The doses (in r.e.f. per a thermal neutron) created by gamma ray and protons in the parts of the body where the dose is a maximum, and average doses throughout the body are calculated. These data permit calculating the maximum permissible fluxes of thermal neutrons on the surface of a body using several assumptions concerning the relative biological effectiveness (RBE) of protons. It is established that if the RBE of protons is taken as 10, the RBE of thermal neutrons in the parts where a maximum dose is created is equal to 2–3 and not 5 as is often assumed. Towards the centre of the body the RBE of thermal neutrons falls and approaches unity, that is, the effect of thermal neutrons on the internal organs should not differ from that of gamma rays.

Since the dose received by a man is determined not by the flux of thermal neutrons in the air, but by the flux on the surface of the body, it is recommended to include the flux of thermal neutrons on the surface of the body in the standards of the maximum permissible radiation levels, the more so as this value is measured by personal dosimeters.

P/2176. Accumulation of radiostrontium and calcium by fresh water fish. HAROLD L. ROSENTHAL (U.S.A.).

The accumulation and deposition of long lived bone seeking radioactive nuclides in animals is of primary interest to those concerned with problems of public health, ecology and radiation hazards. In order to solve these problems, it is necessary to gain an understanding of the metabolism of the radioactive nuclides in question in a variety of animals and plants from different ecological environments. When fresh water fishes (*Lebisles*) are placed in water containing calcium-45 or strontium-90, the fishes take up the isotopes rapidly in a linear fashion with time. The uptake of the isotopes by the organs of the fish is also linear with time. The rate of uptake of calcium-45 or strontium-90 is related to the concentration of inactive calcium or strontium in the water, decreasing in the form of a logarithmic equation as the salt content of the water increases from 0.3 to 20 mM/liter.

The slope of the regression equation varies with the isotope and the amount and kind of salt added to the water and small quantitative differences in the metabolism of strontium and calcium are apparent. In the presence of water containing various amounts of strontium-87, the negative slope for strontium uptake is 0.92 as compared with 0.80 for calcium-45. Conversely, when the two nuclides are added to water containing various amounts of calcium-40, the calcium-45 slope is 0.84 as compared to 0.69 for strontium-90. The rate of uptake of the bone seeking elements is also dependent on the osmotic pressure of the water reaching a maximum rate of calcium-45 when the water contains between 15 and 35 mM/liter of sodium ion. The osmotic effect is considerably smaller than the effect of added calcium or strontium but appears to be important.

The rate of uptake of calcium-45 and strontium-90 from artificial pond water containing no inactive alkaline earth elements (except those of the isotopes used) are the same for both isotopes when expressed as a concentration factor which relates the logarithm of the rate of uptake to the logarithm of the water activity. This is in contrast with the report of Boroughs *et al.*, who showed that salt water fishes (Limn. and Ocean 2: 28, 1957) discriminate against strontium relative to calcium. Discrimination against strontium has also been shown in mammals and plants by many investigators. Although the body of the guppy does not discriminate against strontium relative to calcium, bone and visceral organs accumulate a greater percentage of the body strontium-90 than calcium-45 while muscle accumulates less strontium-90 than calcium-45 when the water in which they swim contains minimal amounts of calcium or strontium. However, when the water calcium-40 is increased to 7 mM/liter the spine : body ratio for the uptake of calcium-45 becomes the same as the spine : body ratio of strontium-90 uptake. On the other hand, the spine : body ratio of strontium-90 remains constant over a wide range of water strontium or calcium concentrations. It would appear, therefore, that fresh water fishes in water of minimal salt concentration are generally calcium deficient and do not discriminate against strontium.

P/2269. Sr^{90} determination in human bones. C. CHAGAS, E. PERALLES PINTO, E. PENNA-FRANCA and A. PINTO COELHO (Brazil).

Following up the work on the determination of Sr^{90} levels obtained in several materials in Brazil, we have done a series of analyses in new-born and still-born children's bones coming from many hospitals of Rio de Janeiro. We used the analytical method employed for the routine work at the Health and Safety Laboratory, which consists of separating Sr from Ca (after bone calcination) by the difference in the solubilities of their nitrates in a 70% nitric acid solution. Radium, a possible contaminating agent, was precipitated from the strontium nitrate salt as Barium chromate. The purified Sr-containing solution was kept in deposit for a minimum of two weeks in order that the Sr^{90}–Y^{90} secular equilibrium was attained. The Y^{90} that grew in the stock solution was separated as an oxalate, set up and counted in a Geiger-Müller counter with low and stable background.

P/2285. Radioisotopes from fusion in rain water: Co^{57} and Mn^{54}. L. MARQUEZ, N. L. COSTA and I. G. ALMEDIA (Brazil).

The 267 day radioisotope Co^{57} has been detected and measured in several rains during one year. The collected

rain water in amount of a few thousand liters was passed through a cation exchange column and then eluted with 4NHCl. Co carrier was added to the rain water before passing it through the column. The Co was submitted to a thorough radiochemical purification and the pure Co was counted with a single channel scintilation spectrometer having a sodium iodide well crystal. The spectra corresponded to that of Co^{57} making the identification of the isotope definite. The spectrometer was calibrated for energy measurements and for the efficiency of the gamma rays.

We have found recently, using a similar technique, that there is also Mn^{54} in the rain water, confirming the results of some American workers.

The presence of Mn^{54} seems to indicate that the bulk of both Mn^{54} and Co^{57} are formed by secondary nuclear reactions in nuclear fusion tests. The activity of Mn^{54} and Co^{57} are now about one order of magnitude smaller than the activity of the fission products Cs^{137} and Sr^{90} which we have measured also in the rain water of Rio de Janeiro.

P/2310. Sorption of tracer amounts of strontium and cesium in soils. V. M. Klechkovsky, G. N. Tselischeva and L. N. Sokolova (U.S.S.R.).

The results of the radioisotope experiments on the processes of adsorption by soils of tracer amounts of strontium and cesium are presented. Data are given on the intensity of adsorption and on the ability to desorption of tracer amounts of sorbed strontium and cesium in different soils because the sorbing properties of soils are an important factor which determines movability of these elements, their capacity of translocating in the soils and of being assimilated by plants. A comparison is made between the role played by calcium of the soil as a "non-isotopic carrier" of radioactive strontium, and the role played by potassium as a "non-isotopic carrier" of radioactive cesium in the sorption process in the soils. On the basis of their investigation the authors cite some figures which characterize specific features of the behavior of tracer amounts of strontium and cesium in different soils.

P/2311. Uptake of strontium, cesium and some other fission products by plants and their accumulation in crop yield. I. V. Gulaykin and E. V. Judintzeva (U.S.S.R.).

The results of investigations conducted by the authors on the uptake of strontium-90, cesium-137, cerium-144, zirconium-95, ruthenium-106 and some other fission products by various agricultural plants, have been summarized.

Data on the distribution of above elements in plant organism and their accumulation in different parts of the crop as a function of biological characteristics of the plant environment conditions have been given.

Possible agrochemical and agrotechnical aspects of the problem of controlling processes responsible for an amount of fission products accumulation and the ratio between strontium and calcium, cesium and potassium in crop are being considered.

The results of the investigations show that the accumulation of radioactive fission products during the vegetative period increases with the growth of the above-ground mass of the plant.

However, the content of fission products per unit of dry substance decreases twice or thrice, as the organic mass of plant grows.

The radioisotopes of strontium and cesium accumulate in the above-ground part of the plant in greater amount than other fission products. Thus, up to 80–90% of the radiostrontium and up to 50–75% of radiocesium absorbed are accumulated in the above-ground parts of the plant; over 90% of other fission products taken up by the plant are accumulated in roots.

The major quantity of the fission products incorporated into the above-ground parts of the plant stores in vegetative organs.

Reproductive organs of plant accumulate radioisotopes of strontium and cesium in greater amount than other fission products.

The takeup of radioactive fission products by plant per unit of dry substance might be higher as compared with the concentration of them in environment.

The cesium-137 content in agricultural plants may also be very high. The coefficient of strontium and cesium content in crop depends on biological characteristics of the plant, soil properties and on other environmental conditions.

P/41. The alpha-ray activity of humans and their environment. W. V. Mayneord, J. M. Radley and R. C. Turner (United Kingdom).

P/60. Some special problems in health physics instrumentation. D. Taylor (United Kingdom).

P/61. Dosimetry aspects of the production of bone tumors by radiation. L. F. Lamerton (United Kingdom).

P/93. Studies on the very early distribution of calcium-45 in the tissues and skeleton of the mouse. K. B. Dawson and E. O. Field (United Kingdom).

P/99. Occupational hazards from exposure to ionizing radiations: a study of the causes of death and the mortality rate among British radiologists. W. M. Court Brown and Richard Doll (United Kingdom).

P/126. Continuous measurement of the radioactivity of atmospheric aerosols. Robert Quivy (Belgium).

P/173. Principles for dealing with safety and location of atomic energy plants in Sweden. T. Magnusson (Sweden).

P/184. Health and safety in Canadian operations. G. C. Butler, C. A. Mawson, J. Neil, C. G. Stewart and G. W. C. Tait (Canada).

P/219. Canadian experience in the measurement and control of radiation hazards in uranium mines and mills. S. D. Simpson, C. G. Stewart, G. R. Yourt and H. Bloy (Canada).

P/235. About the measurement of radiation dosages by thermoluminescence. N. Grögler, F. G. Houtermans and H. Stauffer (Switzerland).

P/236. Special low level counters. J. Geiss, F. G. Houtermans and H. Oeschger (Switzerland).

P/247. Estimation of neutron doses by radiation chemical measurements. W. Minder (Switzerland).

P/248. Fundamental requirements of installations for the continuous monitoring of radioactive-contaminated media. A. Stebler and P. Schranz (Switzerland).

P/261. Evaluation of radiation doses—technique used by the Service de Controle des Irradiations, Geneva. (Switzerland).

P/294. Simultaneous hand and clothing checking for alpha and beta contamination. J. R. Brown (United Kingdom).

P/295. Radiological protection at the Springfields works of the U.K. Atomic Energy Authority. A. Butterworth, A. Cook and H. Mason (United Kingdom).

P/297. The disposal of radioactive liquid wastes into coastal waters. H. J. Dunster (United Kingdom).

P/302. Control of radiation exposure in the U.K. atomic industry. H. J. Dunster, F. R. Farmer, A. S. McLean and W. G. Marley (United Kingdom).

P/308. Development of a self-contained scheme for the treatment of low activity waste. R. H. Burns and E. Glueckauf (United Kingdom).

P/316. District surveys following the Windscale incident. A. S. McLean, H. J. Dunster, H. Howells and W. L. Templeton (United Kingdom).

P/329. Recent instruments for measuring the concentration of radioactive products in the atmosphere. Mme Jehanno and MM. Blanc, Lallemant and Lequais (France).

P/330. Lessons drawn from the study of statistics obtained from individual radiological recording. Dr. Henri Jammet and Dr. François Perrissin (France).

P/331. Various possible ways to express the toxicity of radioactive substances in relation with the involved practical problems. Dr. Henri Jammet and Georges Vacca (France).

P/338. Thoron and hazards associated with the handling of thorium compounds. J. Pradel and F. Billard (France).

P/388. Retention des produits de fission dans la fusion de produits du type vernis céramiques. M. I. Goldman *et al.* (U.S.A.)

P/389. The storage of high level radioactive wastes design and operating experience in the United States. O. H. Pilkey, A. M. Platt and C. A. Rohrmann (U.S.A.).

P/390. Radiation biology research as a supporting function for atomic installations. H. A. Kornberg (U.S.A.).

P/391. Wind pickup of radioactive particles from the ground. J. W. Healy and J. J. Fuquay (U.S.A.).

P/392. Absorption of cesium-137 by components of an aquatic community. Robert C. Pendleton and Wayne C. Hanson (U.S.A.).

P/393. Radioactive materials in aquatic and terrestrial organisms exposed to reactor effluent water under laboratory and natural conditions. J. J. Davies, R. W. Perkins, R. F. Palmer, W. C. Hanson and J. F. Cline (U.S.A.).

P/394. Automatic in-line analyzers for radioactive constituents in liquid and gaseous waste streams. L. C. Schwendiman, H. G. Rieck, J. D. McCormack and R. A. Harvey (U.S.A.).

P/395. The removal of strontium from wastes by a calcite–phosphate reaction. J. R. McHenry, L. L. Ames and J. E. Honstead (U.S.A.).

P/396. Incineration of radioactive wastes. G. W. Morgan (U.S.A.).

P/397. Treatment of radioactive waste gases. A. G. Blasewitz and W. C. Schmidt (U.S.A.).

P/399. Calibration of wild jackrabbits for use in biological monitoring for radioactive iodine. Norman R. French (U.S.A.).

P/400. Some recent developments in reclamation and waste disposal. Lee Gemmell (U.S.A.).

P/404. Weathering of great world soil groups as related to general atmospheric radioactivity. C. Delwiche (U.S.A.).

P/406. Time dependent isotopic variations: a tool in the study of the earth and meteorites. PAUL W. C. GAST (U.S.A.).

P/426. A meteorological study of potential atmospheric contamination from multiple nuclear sites. DONALD H. PACK and CHARLES R. HOSLER (U.S.A.).

P/428. Hazards associated with fission product release. B. P. LEONARD JR. (U.S.A.).

P/429. The evaluation and utilization of meteorology. MAYNARD E. SMITH and IRVING A. SINGER (U.S.A.).

P/432. Modifications, improvements and simplifications of environmental reactor hazard evaluations. J. J. FITZGERALD (U.S.A.).

P/436. A quantitative approach to evaluation of risk in locating a reactor on a given site. HENRY J. GOMBERG, ANN ARBOR, THOMAS BASSETT and CARLOS VELEZ (U.S.A.).

P/437. Reactor containment. R. O. BRITTAN and J. C. HEAP (U.S.A.).

P/438. A numerical method for reactor site evaluation from the hazard-to-people viewpoint. K. W. DOWNES and I. SINGER (U.S.A.).

P/547. Radioactivity in marine organisms. A. H. W. ATEN Jr. (Netherlands).

P/553. Airborne radioactivity after a reactor accident. J. BLOK (Netherlands).

P/554. Low level counting of beta emitters. G. W. BARENDSEN (Netherlands).

P/567. A simple device for calculating the depth dosis by means of photographic recording. DR. KÅRE MYHRE (Norway).

P/577. Health physics work and experience in a small atomic energy establishment. G. JENSSEN (Norway).

P/578. Scattered radiation from a 1700 curies Cs^{137} therapy unit. PER GRANDE and J. L. HAYBITTLE (Norway).

P/586. A method for the determination of very small amounts of I^{131} in biological materials, especially in milk. HELGE BERG (Norway).

P/738. An evaluation of inhalation hazards in the nuclear energy industry. J. N. STANNARD (U.S.A.).

P/740. External environmental radiation measurements in the United States. LEONARD R. SOLON, WAYNE M. LOWDER, ALBERT V. ZILA, HARRIS D. LEVINE, HANSON BLATZ and MERRIL EISENBUD (U.S.A.).

P/741. Radiation protection in the industrial use of radiation. W. A. MCADAMS (U.S.A.).

P/742. Electron stopping powers relative to air. NORMAN A. BAILY and GEORGE C. BROWN (U.S.A.).

P/743. Radiation exposure to people in the environs of a major atomic energy plant. J. W. HEALY, B. V. ANDERSEN, J. K. SOLDAT and H. V. CLUKEY (U.S.A.).

P/745. Radiation protection considerations in a heavy water moderated reactor. C. M. PATTERSON (U.S.A.).

P/746. Analysis of absorbed dose delivered by monoenergetic neutrons. HARALD H. ROSSI and WALTER ROSENZWEIG (U.S.A.).

P/750. Radioactivity standardization in the United States. WILFRID B. MANN (U.S.A.).

P/751. Influence of lowered permissible dose levels on atomic energy operations in the United States. LAURISTON S. TAYLOR (U.S.A.).

P/752. Standards for neutron flux measurement and neutron dosimetry. R. S. CASWELL, J. CHIN and E. R. MOSBURG JR. (U.S.A.).

P/754. Radiation safety experience at the MTR, ICPP, SPERT and ETR. J. W. MCCASLIN and J. P. LYON (U.S.A.).

P/755. Surface dose from plutonium. W. C. ROESCH (U.S.A.).

P/758. An exposure records system for a major atomic energy facility. A. R. KEENE, E. C. WATSON and J. W. VANDERBEEK (U.S.A.).

P/760. Control of health hazards in handling plutonium. DEAN D. MEYER and HARRY F. SCHULTE (U.S.A.).

P/761. Absolute dosimetry of irradiations with charged particles. ROBERT H. SCHULER (U.S.A.).

P/782. Hazards of medical radiation exposure in the nuclear age. WALTER R. STAHL, CLINTON C. POWELL, DONALD R. CHADWICK, EUGENE VAN DER SMISSEN, ARTHUR H. WOLFF and ARTHUR L. KENNEY (U.S.A.).

P/864. Quantitative evaluation of some of the parameters involved in scanning body tissues. RAYMOND L. LIBBY and KATHERINE HAND (U.S.A.).

P/878. Total and partial body radioactivity measurements for metabolic tracer studies. G. J. Hine, J. A. Cardarelli and B. A. Burrows (U.S.A.).

P/888. Hematologic effects in man of low-level radiation exposure. R. Lowry Dobson and Mary M. Chupp (U.S.A.).

P/895. Experimental consideration of the genetic effect of low doses of irradiation on viability in Drosophila melanogaster. A. B. Burdick and T. Mukai (U.S.A.).

P/909. Restoration of X-ray induced damage to the antibody-forming capacity. W. H. Taliaferro and B. N. Jaroslow (U.S.A.).

P/910. Structural changes in human bone containing Ra^{226}. R. E. Rowland, Jenifer Jowsey and John H. Marshall (U.S.A.).

P/970. Chemical dosimetry. Sigurd Rösinger (Federal Republic of Germany).

P/980. The uptake of zinc and strontium in Vitis vinifera. O. Siegel and W. Goerke (Federal Republic of Germany).

P/991. Radiation Sensitiveness of various organs and how this sensitiveness can be influenced by biological radiation protective substances. Ulrich Hagen and Hanna Ernst (Federal Republic of Germany).

P/993. Effects of protective agents applicated after irradiation. Hans A. Künkel and Gerhard Schubert (Federal Republic of Germany).

P/994. Early biochemical reactions following X-irradiation. Heinrich Maass and Gerhard Schubert. (Federal Republic of Germany).

P/999. The influence of meteorological parameters on the propagation of radioactive fission products in the biosphere. Dr. M. Hinzpeter (Federal Republic of Germany).

P/1009. Air leakage from semi-airtight buildings. William K. Luckow and John F. Patterson (U.S.A.).

P/1023. Safeguard considerations in the American nuclear merchant ship program. Richard P. Godwin and Douglas L. Worf (U.S.A.).

P/1028. Methods of evaluating the containment ability of outer containment structures for nuclear reactors. Wilfred E. Baker (U.S.A.).

P/1057. A study of the dynamics of strontium and calcium metabolism and radioelement removal. W. B. Looney, C. J. Maletskos, Marie Helmick, John Reardon, Jonathen Cohen, Warren Guild, John Buchanan, F. I. Visalli and Mary Hawes (U.S.A.).

P/1060. Frequency of bone sarcoma in the United States in relation to low level radiation exposure. John C. Bugher (U.S.A.).

P/1073. Multipurpose processing and ultimate disposal of radioactive wastes. E. G. Struxness and F. L. Culler (U.S.A.).

P/1106. Experience in radiological protection in South Africa. S. F. Ocsthuizen, W. G. Pyne-Mercier, T. Fichardt and D. Savage (Union of South Africa).

P/1138. Regulations concerning radiation protection and survey. MM. Duhamel, Lavie, Fitoussi (France).

P/1175. Disposal of the radioactive effluents at the "Commissariat á l'Energie Atomique". Treatment leading to evacuation into a river. MM. Duhamel, Menoux, Candillon (France).

P/1178. Industrial problems concerning the treatment of radioactive liquid wastes. Solutions retained at Marcoule. C. Bernaud (France).

P/1211. Radiation detection for control and protection. Madame Fuhrmann, MM. Joffre and Savouyaud (France).

P/1238. Safety measures in French uranium mines to ensure staff security against the dangers of radioactivity. Robert Avril, Charles Berger, Francis Duhamel and Jacques Pradel (France).

P/1315. Etude sur l'utilisation d'argiles roumaines comme échangeurs d'ions. G. Ioanid, G. Mastacan, F. Bunus, M. Popesco, P. S. Stanciu et A. Ciocanel (Roumanie).

P/1355. Oceanographical researches on the waste disposal off the coast of Tokai-Mura. (Japan).

P/1384. Alpha air monitory and dosimetry methods. R. Dugnani Lonati (Italy).

P/1467. Body radioactivity measurement as an aid in assessing contamination by radio-nuclides. J. Rundo (United Kingdom).

P/1484. Quantitative and qualitative estimation of scattered radiation from protective barriers. K. A. Mahmoud (United Arab Republic).

P/1515. Elimination de traces d'elements radioactifs dans des solutions aqueuses. M. MILIONE, G. CETINI and F. RICCA (Italy).

P/1516. Inspection and maintenance of large reactors. T. TUOHY (United Kingdom).

P/1551. The experience in the United States with reactor operation and reactor safeguards. C. ROGERS MCCULLOUGH (U.S.A.).

P/1643. A new instrument for the estimation of radon and thoron in air and its applications. K. G. VOHRA (India).

P/1653. Factors affecting the relative biological efficiencies of ionizing radiations. K. C. BORA (India).

P/1659. Experience in radiological protection measures in some Indian medical institutions. P. V. HARIHARAN and P. N. KRISHNAMOORTHY (India).

P/1660. Radiation dose measurements in the monazite areas of Kerala State. D. S. BHARATWAL and G. H. VAZE (India).

P/1676. Treatment of radioactive effluents at the Mol laboratories. P. DEJONGHE (Belgium).

P/1682. Detecteurs de faible activite d'emetteurs de rayonnement alpha et application de ces detecteurs a la mesure de la contamination des eaux, aerosols et surfaces de travail. R. BOULENGER (Belgique).

P/1767. Experience in the disposal of radioactive wastes to the ground. R. E. BROWN, D. W. PEARCE, W. DE LAGUNA, E. G. STRUXNESS, and C. M. PATTERSON (U.S.A.).

P/1768. Fission product recovery from radioactive effluents. R. L. MOORE and R. E. BURNS (U.S.A.).

P/1788. Experience in the dispersal of low level radioactive waste in the Mohawk River. L. J. CHERUBIN, W. H. TRURAN, M. R. KENNEDY and J. J. FITZGERALD (U.S.A.).

P/1796. Recent developments in facilities for handling penetrating radiation. R. C. GOERTZ (U.S.A.).

P/1829. Recent developments in facilities for handling intense alpha sources. W. J. MARAMAN (U.S.A.).

P/1834. Meteorological factors in the appraisal and control of acute exposures to stack effluents. JAMES F. FUQUAY (U.S.A.).

P/1835. Factors affecting the dispersal of fission products in estuarine and inshore environments. D. W. PRITCHARD (U.S.A.).

P/1867. Global scale dispersion by the atmosphere. LESTER MACHTA (U.S.A.).

P/1871. Problems in the disposal of acid aluminum nitrate high-level radioactive waste solutions by injection into deep-lying permeable formations. EDWIN ROEDDER (U.S.A.).

P/1882. Radiation survey methods at UCRL, Berkeley. ROGER WALLACE, BURTON J. MOYER, H. WADE PATTERSON and ALAN R. SMITH (U.S.A.).

P/1913. Feasibility of radioactive waste disposal in shallow sedimentary formations. J. WADE WATKINS, F. E. ARMSTRONG and R. J. HEEMSTRA (U.S.A.).

P/1927. Measurement systems for high level dosimetry. E. J. HART, H. W. KOCH, B. PETREE, J. H. SCHULMAN, S. J. TAIMUTY and H. O. WYCKOFF (U.S.A.).

P/1953. Fission products in the atmospheric precipitation in Debreczen, East Hungary, between 1952-1957. D. BERÉNYI SEN. and A. SZALAY (Hungary).

P/1955. The up-take of radioactive substances through leaves and grass directly from the atmosphere. K. SOMMERMEYER (Federal Republic of Germany).

P/1956. Identification of the components of the atmospheric radioactivity through chemical and radiometric methods. L. ARGIERO, S. MANFREDINI and G. PALMAS (Italy).

P/1957. Radioactivite des pluies du 4 au 14 Novembre 1957 dans le District de Lisbonne. BRANCA EDMÉE MARQUES et MARIA REGINA GRADE (Portugal).

P/1958. Fallout measurements in Cracow in 1957. L. JURKIEWICZ and M. MASSALSKA (Poland).

P/1959. Fallout in cistern water. A. GRÜTTER, F. G. HOUTERMANS, C. MÜHLEMANN and H. OESCHGER (Switzerland).

P/1963. New measurements on the world-wide distribution of natural and artificially produced tritium. F. BEGEMANN (Federal Republic of Germany).

P/1995. Native weeds as indicators of radioactive contamination of the soil. WADE T. BATSON, WILLIAM R. KELLEY and WILLIAM E. HOY (U.S.A.).

P/1996. The metabolism of Sr^{90} and Y^{90} and the influence of lactation on retention. FRANK E. HOECKER and EDWARD I. SHAW (U.S.A.).

P/2004. Handling, treatment, and disposal of radioactive wastes—An assessment. JOSEPH A. LIEBERMAN (U.S.A.).

P/2024. Desactivation of diluted salt solutions and low active sewage of radiochemical plants. G. A. SEREDA *et al.* (U.S.S.R.).

P/2025. Methods for removal of radioactive substances from sewage of research institutions. K. A. Dolchakov and F. V. Raouzen *et al.* (U.S.S.R.).

P/2077. Long-term effects produced by small doses of radioactive substances in chronical experiment. N. A. Kraievsky, D. I. Zakutinsky, E. B. Kurlyandskaya, Y. I. Moskalev, V. N. Streltsova, L. N. Burykina, N. N. Litvinov and Y. N. Soloviov (U.S.S.R.).

P/2106. Rapid assessment of the surface radioactive contamination. F. Béhounek (Czechoslovakia).

P/2130. Measurement and analysis of the air contaminated with low alpha-active aerosols. V. I. Polikarpov *et al.* (U.S.S.R.).

P/2171. Procedures and criteria in the location and safety evaluation of reactors. Clifford K. Beck (U.S.A.).

P/2207. Migration of radioelements in soils. V. I. Spitsyn, V. D. Balukova, A. F. Naumova, G. I. Grafov *et al.* (U.S.S.R.).

P/2267. On the nature of long range fallout particles. B. Gross and F. X. Roser (Brazil).

P/2268. Long range fallout activity in Rio de Janeiro 1956–1958. A. Aron, E. Meyer and B. Gross (Brazil).

P/2275. Radiation intensity levels and air and water activities observed with the IEAR-1 swimming pool reactor at 5 MW. R. R. Pieroni, S. B. Herdade, Wilma S. Hehl and C. R. Pereira (Brazil).

P/2284. On the up-take of MsTh$_1$ in naturally contaminated areas. F. X. Roser (Brazil).

P/2287. Argonne automatic data processing system. H. Moses and F. C. Kulhanek (U.S.A.).

P/543. Hot laboratory facility for physical measurements on irradiated plutonium. R. C. Goertz

P/1249. Methodes de prospection autoportee et aerportee la technique de la detection des rayonnements les perspectives offertes par la discrimination des energies. J. Berbezier, B. Blangy, J. Guitton et C. Lallement.

P/1658. Factors affecting the relative biological efficiencies of ionizing radiations. K. C. Bora.

P/1898. Mapping the distribution of gamma ray emitting isotopes with the scintillation camera. H. O. Anger.

P/2083. An instrument for measuring low fluxes of high energy neutrons. P. S. Baranov, V. I. Goldanskii and V. S. Roganov.

P/2107. Plastic scintillators and some aspects influencing their luminescent yield. J. Blecha and J. Silar (Czechoslovakia).

P/2288. Progress in application of meteorology to peaceful uses of atomic energy. J. Z. Holland (U.S.A.).

P/2341. Characteristic energy absorption constant and its uses in radium and isotope dosimetry. K. C. Tsien (China).

P/2343. Compact low level counting arrangement. K. van Duuren, W. K. Hofker and J. Hermsen (Netherlands).

P/2351. Disposal of high level radioactive liquid wastes in terrestrial pits—a sequel. W. de Laguna, K. E. Cowser, F. L. Parker and (U.S.A.).

P/2352. Legal problems of liability and financial protection connected with radiation injuries. E. Blythe Stason (U.S.A.).

P/2354. The treatment of large volume, low level waste by the lime-soda softening process. K. E. Cowser, R. J. Morton and E. J. Witkowski (U.S.A.). (Prepared by: E. G. Struxness and K. E. Cowser.)

P/2359. Disposal of radioactive wastes from the uranium ore refining industry. E. C. Tsivoglou (U.S.A.).

P/2382. Evaluation of internal exposure hazards for several radioisotopes encountered in reactor operations. J. E. Ballou, W. J. Bair, L. K. Bustad, L. A. George, P. L. Hackett, S. Marks, V. H. Smith, M. F. Sullivan, L. A. Temple, R. C. Thompson and D. H. Willard (U.S.A.). (Prepared by R. C. Thompson, W. J. Bair, S. Marks and M. F. Sullivan.)

P/2397. Calcium and strontium uptake and metabolism. W. F. Neuman (U.S.A.).

P/2407. Reactor safety, hazards evaluation and inspection. C. K. Beck, M. M. Mann and P. A. Morris (U.S.A.).

P/2108. Measurement of absolute dosis of gamma-radiation with metallic ionization chambers. J. Klumpar (Czechoslovakia).

P/2110. A reproducible chemical dosimeter for the measurement of low doses of x- and gamma-rays. Z. Spurny (Czechoslovakia).

P/2172. The effects of acute and chronic exposure of plants to ionizing radiations. Arnold H. Sparrow (U.S.A.).

P/2128. Atmospheric radiocarbon as a tracer in geophysical circulation problems. T. A. Rafter and G. J. Ferguson.

P/2243. Program of the international commission on radiological units and measurements. L. H. Gray, L. S. Taylor and H. O. Wyckoff (WHO).

P/2245. Gamma-emanation method of evaluation of radioactivity anomalies. Y. P. Bulashevich (U.S.S.R.).

Health Physics Pergamon Press 1958. Vol. 1. pp. 255–267. Printed in Northern Ireland

RADIATION EXPOSURES FROM NUCLEAR TESTS AT THE NEVADA TEST SITE

G. M. DUNNING
Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington, D.C.

(*Received* 29 *May* 1958)

Abstract—The purpose of this article is to summarize some of the pertinent data on radiation exposures around the Nevada Test Site as a result of nuclear tests. The data are categorized into:

(I) External γ-radiation.
(II) Activity concentrations in air.
(III) Activity concentrations in water.

Under each of these areas there is considered:

(A) The methods used to estimate the radiation exposure.
(B) The relevant data collected.
(C) Evaluation of the data.

The maximum external exposure has been up to 7–8 r for the fifteen persons involved. In terms of relative large populations the average exposure for the 1,000,000 persons living nearest the Test Site has been at the rate of about $\frac{1}{2}$ r/30 years. The highest concentration of fallout activity in the air has been about 1.3 $\mu c/m^3$ averaged over the 30 hr that the activity was present in significant quantities. The highest concentration of fallout activity in a potential drinking water supply has been about 1.4×10^{-4} μc/ml extrapolated to $D + 3$ days. Some evaluation is given for these data in the article.

I. EXTERNAL GAMMA RADIATION

A. *Methods of estimating*

The method used to compute the "estimated" γ-dose in communities around the Nevada Test Site is described in the first reference.[1] These calculations involve the parameters of radiological decay, weathering of the activity and shielding afforded personnel by normal structures. None of these, especially the latter two, will be identical for all cases.

The decay of activity of mixed fission products is generally assumed to follow the relationship of $(\text{time})^{-1.2}$. This is intended to apply to disintegrations of atoms. In estimating the reduction in time of γ-dose rates above a fallout field, there should be considered the changing numbers and energy spectra of γ-photons released per disintegration (as well as any effects of weathering). This estimation has been attempted.[2,3] These analyses indicate that when computing total doses which accrue from a fallout of fission products occurring within a few hours after detonation, the integration of the $(\text{time})^{-1.2}$ curve gives a fair approximation since most of the dose is accumulated during the early periods.

The effects of weathering (winds and rains) upon dose rates obviously will vary depending upon the time of occurrence and the degree of movement of the fallout material. In those localities where only one or a few measurements are made in time, a generalized rule for the effects of weathering must be assumed. For these cases, it has been assumed by us that the γ-dose rates at 3 ft above the ground will decrease according to $(\text{time})^{-1.2}$ for the first week after a detonation, $(\text{time})^{-1.3}$ for the second week and $(\text{time})^{-1.4}$ thereafter.

The protection afforded personnel by normal structures will also vary according to the type of structures and time spent within them. As a result of measurements taken within homes around the Nevada Test Site it was estimated that the attenuation might be about a factor of two. Assuming about half of the

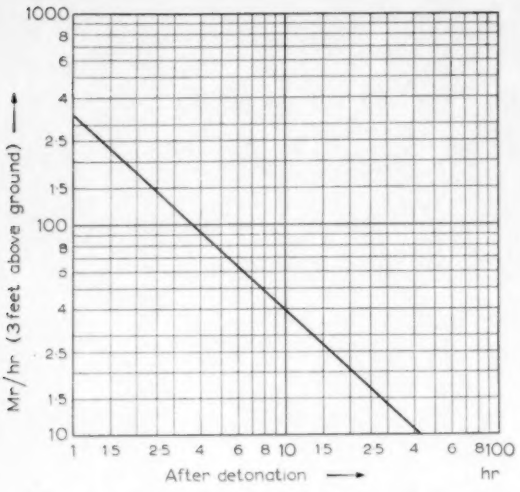


FIG. 1. Approximate γ-dose rates vs. time after detonation to produce 1 r estimated dose.

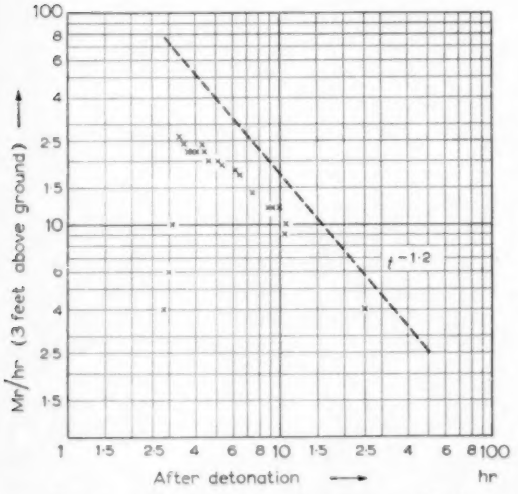


FIG. 2. γ-Dose rates at St. George, Utah (17 March 1953).

time is spent indoors, this means a 25 per cent reduction per day of the out-of-doors dose.

These assumed effects of radiological decay, weathering and shielding are combined into the preparation of Fig. 1.

Fallout of radioactive debris does not occur instantaneously. Fig. 2 shows the effects of the combination of fallout occurring in time, together with the concurrent radiological decay, at St. George, Utah on 17 March 1953. In estimating doses to communities around the Nevada Test Site, based only on monitoring data obtained after the peak has occurred, it has been assumed that all of the fallout occurred near the time of initial appearance. In this respect these "estimated" doses tend to overestimate the actual dose.

In any given community usually there will be an initial variation of dose rates owing to a somewhat uneven distribution of fallout. The higher values are used in computing the "estimated" dose.

B. *The data on external γ-exposures*

Table 1 summarizes the "estimated" doses to communities around the Nevada Test Site based on the above methods, as well as data obtained from the use of film badges and automatically recording instruments.[4]

The large mass of data collected over the past 7 years is currently being cross-checked and re-evaluated by a committee headed by Dr. A. VAY SHELTON of the University of California Radiation Laboratory. The analyses are still too preliminary to report results at this time.

C. *Evaluation of the external γ-exposures*

In June 1956 the National Academy of Sciences stated "... that individual persons receive not more than a total accumulated dose to the reproductive cells of 50 r up to age 30 years—and not more than 50 r additional up to age 40 ..." and "... that for the present it be accepted as a uniform national standard that X-ray installations (medical and non-medical), power installations, disposal of radioactive wastes, experimental installations, testing of weapons and all other humanly controlled sources of radiations be so restricted that members of our general population shall not receive from such sources an average of more than 10 r, in addition to background, of ionizing radiation as a total accumulated dose to the reproductive cells from conception to age 30."[5]

On 8 January 1957 the National Committee on Radiation Protection recommended "The maximum permissible dose to the gonads for

Table 1

Community	Approximate population	External γ-radiation (r) Operation Plumbbob	Cumulative for all nuclear tests
Arizona			
Beaver Dam Lodge			
Between Littlefield, Arizona and Utah			
Line, on Highway 91	6	0.21	2.21
Littlefield	44	0.25	0.25*
Mt. Trumbull	30	0†	0.16
Short Creek	90	0	1.60
Wolf Hole	10	0	1.30
California			
Quincy	1330	0.10	0.10
Nevada			
A and B Mine			
(South of M and M Mine)	4	0.10	0.10
Acoma			
(23 miles SE of Caliente)	16	0	3.00
Alamo	250	0.02	1.32
Apex	15	0	0.10
Ash Springs	5	0.01	0.61
Austin	520	0.14	0.19
Baker	60	0.10	0.90
Barclay (22 miles SE of Caliente)	38	0	2.00
Bardoli Ranch			
8 miles E of Nyala	4	0.28	0.28
Basalt	5	0.17	0.17
Belew Ranch			
Located at old Adaven Post Office	550	0.10	0.15
Beatty	3	0.30	0.30
Belmont	6	0.70	0.70
Buckhorn Ranch	12	0.04	0.94
Bunkerville	250	0	4.30
Butler Ranch			
30 miles S of Alamo	1	3.22	3.22
Caliente	970	0	0.70
Carp			
On UP, RR, 25 miles S of Caliente	Variable to 25	0.20	3.80
Charleston Lodge	60	0	0.15
Clarke's Station			
32 miles E of Tonopah on Highway no. 6	Variable 0–5	0	0.80
Coaldale	25	0.45	0.45
Crestline	22	0	0.70
Crystal	5	0	4.00
Crystal Springs	5	0.08	1.08

* Indicates relatively little fallout prior to Operation Plumbbob.
† Zero means no substantial fallout.

Table 1—(*continued*)

Community	Approximate population	External γ-radiation (r) Operation Plumbbob	Cumulative for all nuclear tests
Nevada (*continued*)			
Currant	75	0.27	0.77
Dodge Construction Co.	175	2.2	2.20
Dry Lake	20	0	1.00
Duckwater	50	0.02	0.82
East Ely	1000	0	0.60
Eden Creek Ranch 22½ miles NW of Highway 25 and 64 miles SW of Crystal Springs	0	0	0.70
El Dorado	3	0.36	0.36
Elgin	Variable, 30 avg.	0.03	3.53
Ely	3558	0.43	1.03
Eureka	500	0.4	0.60
Fallini Ranch	15	0.78	1.58
Fallon	2400	0.13	0.13
Fernley	418	0.1	0.10
Gault	0	2.86	2.86
Geyser Ranch	5	0.37	0.37
Glendale	50 avg.	0.08	0.78
Goldfield	220	0.52	0.52
Gold Point	10	0.17	0.17
Groom Mine	0	0.02	2.02
Hawthorne	1861	0.12	0.12
Hiko	55	0.05	1.05
Hoya	8	1.40	2.80
Indian Springs	250	0.09	0.14
Kimberly	120	0.05	0.55
Las Vegas	47,000	0	0.20
Lida	25	0.19	0.19
Lincoln Mine	250–500	1.25	5.25
Lockes	4	0.32	1.62
Logandale	300	0.03	0.43
Lund	250	0.28	0.36
Manhattan	42	0.14	0.14
M and M Mine (8 miles W of Warm Springs (on Highway 6) and 8 miles N of Highway 6)	2	2.34	2.34
Mesquite	590	0.08	1.88
McGill	2300	0.05	0.45
Millett (Ranch only—on Highway 8A, 33 miles S of Highway 50)	5	0.32	0.32
Mina	450	0.13	0.13
Moapa area	175	0.07	0.87
Montgomery Pass	3	0.15	0.15

Table 1—(*continued*)

Community	Approximate population	External γ-radiation (r) Operation Plumbbob	Cumulative for all nuclear tests
Nevada (*continued*)			
North Las Vegas	13,000	0	0.20
Nyala (Between Warm Springs and Adaven (H.N. Sharp Ranch))	6	0.17	1.87
Overton	725	0.01	0.36
Pahrump	89	0	0.20
Panaca	500	0.03	0.68
Parmans Ranch (10 miles S of Belmont)	7	0.78	0.78
Pioche	1392	0.03	0.73
Preston	60	0.31	1.01
Rattlesnake Maintenance Station	4	0.24	0.24
Reed	0	0	4.00
Reveille Mills (About 10 miles SE of Warm Springs (Highway 6))	4	3.40	3.40
Riverside Cabins	15	0	7–8
Round Mountain	200	0.17	0.22
Rox	13	0.21	3.21
Ruby Hill Mine	50	0.40	0.40
Ruth	1244	0	0.50
Schurz	100	0.13	0.13
Sharps (Adaven)	25	0	1.20
Shoshone	250	0	0.70
Silver Peak	7	0.38	0.38
Stone Cabin Ranch 23 miles E of Tonapah and 6 miles N of Highway 6	8	0.42	0.42
Stonehouse Ranch 5 miles N of Belmont on Nev. no. 82	4	0.27	1.27
South Paw Mine 5 miles N of Mt. Irish	3	0.41	0.41
Sunnyside	26	0.27	1.47
Tonapah	1375	0.62	0.62
Uhaldi Ranch About 1 mile S of Old Adaven P.O. location	5	0.29	0.29
Ursine	25	0	0.60
Walch Pine Creek Ranch	4	0.21	0.21
Warm Springs on Highway 6	55	0.25	0.75
Warm Springs Ranch 4 miles NW of Moapa and SW of Highway 93	58	0.04	1.04

Table 1—(*continued*)

Community	Approximate population	External γ-radiation (r) operation Plumbbob	Cumulative for all nuclear tests
Utah			
Alton	154	0.02	0.82
Anderson Junction	17	0.40	1.60
Bear Valley Junction (Orton Junction)	15	0	0.40
Beaver	1685	0	0.25
Beryl Junction	8	0.03	1.03
Cedar City	6106	0.12	0.52
Central	49	0.30	0.30
Enterprise	800	0.05	0.75
Garrison	125	0.10	0.80
Glendale	275	0.06	1.26
Gunlock	127	0.24	2.24
Hamilton Fort	26	0.12	0.72
Harrisburg Junction	5	0.11	0.11
Hurricane	1375	0.08	4.28
Ivins	75	0.50	0.50
Kanab	1900	0	1.60
Kanarraville	263	0.47	1.67
Leeds	100	0.54	3.54
Long Valley	10	0.01	0.81
Lund	75	0	0.50
Milford	1673	0	0.10
Minersville	593	0	0.20
Modena	100	0.02	0.52
Mt. Carmel	125	0.02	0.87
New Castle	115	0.04	0.64
New Harmony	126	0.04	1.24
Orderville	371	0.05	1.55
Panguitch	1501	0.36	0.56
Paragonah	404	0.01	0.41
Parowan	1455	0.01	0.41
Pintura	50	0.62	1.82
Rockville	125	0.07	3.07
St. George	5000	0.50	3.50
Santa Clara	319	0.46	3.96
Shivwits	95	0.54	3.34
Springdale	209	0.05	2.65
Toquerville	219	0.17	2.17
Veyo	100	0.65	2.65
Virgin	135	0.08	1.58
Washington	435	0.13	3.13
Zane	25	0	0.30

the population of the United States as a whole from all sources of radiation, including medical and other manmade sources, and background, shall not exceed 14 million rems per million of population over the period from conception up to age 30, and one-third that amount in each decade thereafter. Averaging should be done for the population group in which cross breeding may be expected." Also, "while at the moment it is not feasible to determine the average exposure for the population with any reasonable accuracy, the adoption of some figure is necessary for planning purposes. For the immediate future, it may be assumed that the total integrated RBE dose received by all radiation workers will be small in comparison with the integrated RBE dose of the whole population. Furthermore, persons outside of controlled areas, but exposed to radiation from a controlled area, constitute only a small portion of the whole population. Therefore, if this small portion is assumed to receive yearly an average per capita dose of 0.5 rem, the total dose to the whole population from manmade radiations is not likely to exceed 10 million rems per million of population up to age 30. (This assumes a dose of 4 million rems per million of population over this age period from background radiation.)"[6]

Whereas, it may be possible to control the small incremental accumulation of an individual's radiation exposure during normal activities within an atomic energy installation, by the very nature of fallout phenomena one community may receive a relatively high exposure from a single fallout. Thus the Atomic Energy Commission has adopted for populations off-site, as an operational guide a maximum of 3.9 r per test series at the Nevada Test Site. Of course, extensive efforts are made to reduce the fallout to a minimum.[7]

The exposures to relatively small numbers of persons (in the context used here) are reported in Table 1. For larger populations, it was estimated that the one million persons living nearest to the Nevada Test Site have received an average per capita exposure of about 0.1 r for the first 6 years of testing at the Nevada Test Site. This is at a rate of about a $\frac{1}{2}$ r/30 years.[8] Operation Plumbbob (1957) contributed only a small increment to the average per capita exposure for these one million persons, and since this value now encompasses 7 years of testing, the rate of about $\frac{1}{2}$ r/30 years remains essentially unchanged.

II. ACTIVITY CONCENTRATIONS IN AIR

A. *Method of estimating radiation exposure from this source*

(1) "Infinity" doses to the lungs. *Assumptions:*

(a) The rate of inhalation is 20 m^3/24 hr,* (0·83 m^3/hr).

(b) The percentage of initial retention of particles (and activity) is 25 per cent.

(c) All of the air-borne activity is associated with particles in the respirable range.†

(d) The mass of the lungs is 1000 g and is uniformly irradiated.‡

(e) Mean energy of the β-particles = 0·4 MeV. (The relative dose from γ-emission may be roughly 10 per cent of the β-dose. Since this is less than the uncertainties in other estimations described below, it will not be considered here.)

(f) The dose rate decreases according to the relationship of $(\text{time})^{-1.2}$.

Step 1. Calculate the dose rate for 1.0 $\mu c/m^3$ per hr.

(a) For each 1 $\mu c/m^3$ activity present in the air per hour, (0.83) (0.25) = 0.21 μc of activity are initially retained.

(b) The initial dose rate to the lungs will be:

$$\frac{(0.21)(\mu c)(1.3 \times 10^8)(\text{d/hr per } \mu c)\ (0.4)(\text{MeV}) \times (1.6 \times 10^{-6})\ (\text{ergs/MeV})}{100\ (\text{ergs/g per rad})(1000)\ (\text{g})}$$

$$\simeq 1.7 \times 10^{-4}\ \text{rads/hr}$$

* There will be higher volumes of intake during working hours than at other times. When such times are known, the assumption usually made is that 10 m^3 are inhaled during 8 hr of work and 10 m^3 inhaled for the remainder of the day.

† Generally, the cascade impactor data have shown 80–90+ per cent of the activity associated with particles 5 μ or less in size. However, it is quite possible that the high volume air samplers upon which most of the data are based, collect larger size particles which would tend to overemphasize the estimated lung dose.

‡ These may tend to underestimate the dose.

Step 2. Calculate the "infinity" dose to the lungs.

(a) $D_\infty = 5A_t t$

A_t = dose rate at time of deposition t after detonation (hr)

t = time after detonation that deposition occurs.

D_∞ = infinity dose (rads)

(b) $D_\infty = (5)(1.74 \times 10^{-4})(t)$ rads

$= (0.85)(t)$ mrads (per 1.0 $\mu c/m^3$ per hr)

[Based on 20 m^3/24 hr]

or $D_\infty = (1.28)(t)$(per 1.0 $\mu c/m^3$ per hr)

[Based on 10 m^3/8 hr]

$D_\infty = (0.64)(t)$(per 1.0 $\mu c/m^3$ per hr)

[Based on 10 m^3/16 hr]

(2) Doses to the thyroid. The method of estimating doses to the thyroid from inhalation of fallout material containing radioactive isotopes of iodine is described in the ninth reference.[9]

B. *The data on concentrations in air*

Ranger (spring 1951).[10] The highest average concentration of fallout activity measured off-site was about 1.7×10^{-5} $\mu c/m^3$ for the $2\frac{3}{4}$ hr the samplers were in operation at Charleston Peak, Nevada. Of course other areas not monitored, may have shown greater concentrations but since all five detonations during Operation Ranger were air bursts, it is unlikely that very high values were present off-site.

Buster-Jangle (autumn 1951).[11] The highest measured concentration of fallout debris in air was at Ely, Nevada, amounting to 0.202 $\mu c/m^3$ averaged over 24 hr. All other measurements were in order of magnitude lower.

Teapot (spring 1955).[14] "... The highest concentration of radioactivity in the air following any one detonation was at Ely, Nevada. This amounted to about $\frac{6}{1000}$ $\mu c/m^3$ averaged over the 28 hr that the material was present in significant quantities ...."

Plumbbob (spring–summer 1957).[15] The measurements of concentrations of fallout material activities in the air during Operation Plumbbob and calculations of radiation exposures to the lungs show that the highest total accumulated dose (recorded at Eureka, Nevada, with a population of about 500) was less than that to be expected from breathing air containing normal amounts of naturally radioactive materials, for a period of about 2 weeks.

Tumbler-Snapper (spring 1952).[12]

Detonation	Locality	24 hr average concentration ($\mu c/m^3$)
7 May 1952	Ogden, Utah	0.020
25 May 1952	Price, Utah	0.001
1 June 1952	Elko, Nevada	0.024
5 June 1952	Elko, Nevada	0.014

Upshot-Knothole (spring 1953)[13]

Locality	24 hr average concentration ($\mu c/m^3$)
St. George, Utah	1.29
Lincoln Mine, Nevada	4.0×10^{-1}
Mesquite, Nevada	1.7×10^{-1}
Groom Mine, Nevada	3.4×10^{-2}
Pioche, Nevada	2.0×10^{-2}
Nellis AF Base, Nevada	1.7×10^{-2}
Ely, Nevada	1.6×10^{-2}
Las Vegas, Nevada	1.0×10^{-2}

C. *Evaluation of activity concentrations in air*

The method of estimating doses to the lungs suggested above clearly shows that in addition to the specific air concentration, the age of the fallout material is an important factor. A single value for a 24 hr concentration gives some perspective as to the potential dose but a more quantitative evaluation requires further

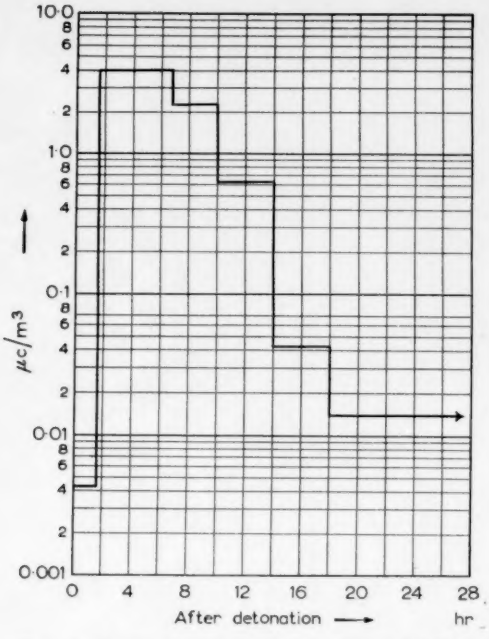


FIG. 3. Approximate concentrations of fallout activity in the air at St. George, Utah (19–20 May 1953).

analysis of the data. This is done below for the activity in the air at St. George, Utah on 19–20 May, 1953 which resulted in the highest estimated lung dose from any single event or in fact, from all fallouts in any given community.

Fig. 3 shows the approximate air concentrations at St. George, Utah 19–20 May, 1953 and a summary of the calculations based on these data are given below:

The above estimates are based on infinity doses. The actual dose will be less than this but the exact value is difficult to estimate. It is probable that some 40 per cent of the material (and activity) will be cleared from the lungs with a half-time of about 1 day. The remainder may have a half-time of removal of about 180 days.[16] Fallout that occurs 1 hr after a detonation will expend about 50 per cent of the potential infinity dose during the first day, while fallout that occurs 10 hr after a detonation requires about 15 days to deliver one-half of the infinity dose. The uncertainties in these and other estimates and assumptions do not justify attempting an exact calculation of actual radiation lung dose. However, it seems reasonable to assume the actual dose is less than the infinity dose and more than one-half of the maximum value.

The biological significance of doses to the lungs may be appraised by comparison with lung doses produced by natural occurring radioactive materials in the air (radon and thoron and their daughter products). These dose estimates vary owing to different assumptions as to natural air concentrations, percentages of activity initially retained in the lungs, rate of elimination, weight and volume of the lung tissue irradiated and the relative biological effectiveness of the α-particles. Some estimates made in this country are 2·6–26 mrems,[17] 70 mrems,[18] 50–100[19] and 100 mrems[20] per week. Based on breathing continuously radon and radon daughter products at a radon level of 1×10^{-11} c/l. (*not* assumed to be a normal concentration in air), HOLADAY

Air concentrations ($\mu c/m^3$)	Midpoint of sampling period (hours after detonation)	Duration of sampling	Infinity lung dose (mrems)
4.0	4.5	5	115*
2.3	8.5	3	75*
0.62	12.0	4	19†
0.43	16.0	4	18†
0.014	24.0	12	2.6†

* Based on 10 m³/8 hr.
† Based on 10 m³/16 hr.

et al.[21] have estimated an average dose from 190 to 43 mrems/week to the lung and from 2160 to 720 mrems/week to the bronchi depending upon the assumptions used. HULTQUIST[22] estimates values from 0·016 mrem/week for the lower limit within a ventilated wood structure to 4000 mrems/week as an upper limit within a lightweight concrete structure with alum shale in Sweden.

In evaluating doses to the lungs from inhalation of fallout, we have been using a comparative value of roughly 50 mrems/week from natural activity. Thus, the lung dose calculated above for fallout at St. George, Utah may be equivalent to that for a greater time period than we originally estimated. "... The estimated radiation dose to the lungs resulting from this activity was about the same as would be expected from breathing air containing normal amounts of radioactive substances for a period of less than a month."[23]

The doses to the thyroid from the same fallout at St. George were estimated to be about 0.3 rep according to the ninth reference.[9] These original calculations were based on an average inhalation of 0.83 m^3/hr. If the same rates of respiration are assumed as for estimating lung doses above, the value would be about 0.4 rep. Many thousands of rep doses are delivered to the thyroid during treatment of disorders of this gland without observable detrimental effects.[24, 25] There is some evidence that as little as 200 reps to the infant thyroid may produce carcinoma in a few percent so exposed.[26,27]

III. ACTIVITY CONCENTRATIONS IN WATER

A. *Methods of estimating internal doses*

The method of estimating doses to the gastrointestinal tract, thyroid and bones resulting from ingestion of water containing radioactive fallout are contained in the twenty-eighth reference.[28] An additional consideration for estimating doses to the bones is given below. *Assumptions:*

(1) The gross fission product activity is 5×10^{-3} μc/ml on $D + 3$ days.

(2) The daily intake is 2200 ml (probably an overestimation).

(3) This is the sole supply of water for 70 years.

Step 1. Compute the total dose to an organ from a daily intake of constant volume or mass but with the activity intake decreasing according to radiological decay of the isotope.

(a) The dose (ignoring for the time any biological decay) to the organ from any day's intake is:

$$D_1 = R_0 \int_0^T \exp(-\lambda_r \tau) d\tau - R_0 \int_0^\tau \exp(-\lambda_r \tau)\, d\tau$$

where

D_1 = dose from any day's intake
R_0 = initial daily dose rate
λ_r = radiological decay
T = number of days intake
τ = time in days (variable)

$$D_1 = \frac{R_0}{\lambda_r}(\exp(-\lambda_r \tau) - \exp(-\lambda_r T))$$

(b) The total dose (ignoring biological decay) to the organ is:

$$D_T = \frac{R_0}{\lambda_r} \int_0^T (\exp(-\lambda_r \tau) - \exp(-\lambda_r T)) d\tau$$

$$D_T = \frac{R_0}{(\lambda_r)^2}[1 - \exp(-\lambda_r T) - \lambda_r T \exp(-\lambda_r T)]$$

(c) The dose to the organ considering biological decay is:

$$D_T = \left[\frac{\lambda_r}{\lambda_r + \lambda_b}\right] \frac{R_0}{(\lambda_r)^2} \times [1 - \exp(-\lambda_r T) - \lambda_r T \exp(-\lambda_r T)]$$

$$D_T = \frac{R_0}{(\lambda_r)(\lambda_r + \lambda_b)} \times [1 - \exp(-\lambda_r T) - \lambda_r T \exp(-\lambda_r T)]$$

Step 2. Compute initial daily dose rate (R_0).

$(5 \times 10^{-3})(2200) = 11$ μc gross fission product intake on $D + 3$

$(11)(5.8 \times 10^{-5}) = 6{\cdot}37 \times 10^{-4}$ μc of Sr^{90} intake on $D + 3$*

* Under the assumptions used here, the strontium-90 content accounts for almost all of the total dose. Percentages of strontium-90 are based on references 29 and 30.

$(6.37 \times 10^{-4})(0.25) = 1.59 \times 10^{-4}$ μc of Sr^{90} deposited in the bones†

$$\frac{(1.59 \times 10^{-4})(\mu c)(3.2 \times 10^{9})(\text{d/day per } \mu c) \times (1.0)(\text{MeV})(1.6 \times 10^{-6})(\text{ergs/MeV})}{100(\text{ergs/g per rad})(7 \times 10^{3})(\text{g})} = 1.16 \times 10^{-6} \text{ rads/day}$$

Step 3. Compute the total dose for 70 years (D_T).

$$D_T = \frac{R_0}{(\lambda_r)(\lambda_r + \lambda_b)} \times [1 - \exp(-\lambda_r T) - \lambda_r T \exp(-\lambda_r T)]$$

where

$R_0 = 1.16 \times 10^{-6}$ rad/day
$\lambda_r = 6.86 \times 10^{-5}$ day^{-1}
$\lambda_b = 1.78 \times 10^{-4}$ day^{-1}
$T = 25{,}550$ days

$$D_T = \frac{1.16 \times 10^{-6}}{(6.86 \times 10^{-5})(2.47 \times 10^{-4})} \times [1 - \exp\{-(6.86 \times 10^{-5})(25{,}550)\} - (6.86 \times 10^{-5})(25{,}550 \exp\{-(6.68 \times 10^{-5}) \times (25{,}550)\}] \cong 35.8 \text{ rads}$$

B. *The data on concentrations in water*

Ranger (spring 1951).[31] No water samples were taken during Operation Ranger. However, the fact that all five detonations were air bursts, and appreciable fallout outside the Test Site was not observed, leads one to estimate that any possible water contamination was quite small.

Buster-Jangle (autumn 1951).[32] Six samples from Lake Meade showed negative results.

Plumbbob (spring–summer 1957).[36] "It was difficult to extrapolate precisely all of the measurements taken of activity in the water to 3 days after the detonation during Operation Plumbbob. This was because after the second shot of the series it was possible that the measured activity represented fallout from more than one

Tumbler-Snapper (spring 1952).[33]

Date	Analysis (μc/ml at 3 days after detonation)
1 May 1952 Crystal Springs Pond	0.5×10^{-8}
1 May 1952 Pahranagat Lake	1.0×10^{-8}
2 May 1952 Caliente (drinking water)	0.28×10^{-8}
2 May 1952 Creek North of Caliente	1.1×10^{-8}

Upshot-Knothole (spring 1953).[34]

Locality	Concentration (μc/ml) extrapolated to 3 days after detonation)
Virgin River Irrigation Canal, Nevada	8.7×10^{-5}
Irrigation Ditch, 56 miles north of Pioche, Nevada	4.5×10^{-5}
Lower Pahranagat Lake, Nevada	3.2×10^{-6}
Virgin River at Mesquite, Nevada	2.6×10^{-6}
Bunkerville, Nevada (tap water)	1.2×10^{-6}
Crystal Springs, Nevada (tap water)	1.1×10^{-6}

Teapot (spring 1955).[35]

Locality	Concentration (μc/ml extrapolated to 3 days after detonation)
Upper Pahranagat Lake, Nevada	1.4×10^{-4}
Waterhole near State (Nevada) Highway 25	9.2×10^{-5}
Meadow Valley (Nevada) Wash	3.3×10^{-5}

† It is recognized that the amount of strontium-90 absorbed into the body is strongly a function of the calcium content of the food and water supply. These estimates are intended only to give perspective to the problem, not to be precisely quantitative.

explosion. The data do show that even the highest activity observed in a drinking water supply (Fallini Ranch, Nevada–population about 15) probably was less than one-hundredth of the operational guide...."

C. *Evaluation of activity concentrations in water*

The highest activity measured in any potential drinking water supply was at Upper Pahranagat Lake, Nevada during the 1955 test series amounting to 1.4×10^{-4} μc/ml extrapolated to 3 days after the detonation. (The Atomic Energy Commission has adopted as an operational guide a maximum of 5×10^{-3} μc/ml of gross fission products, extrapolated to $D + 3$ days). If this water from Upper Pahranagat Lake had been stored and used as a sole supply for 70 years the total internal doses would be estimated as follows:

Bones	~1.0 rad
Lower large intestine (with less doses to other parts of the G.I. tract according to reference 28)	~0.25 rad
Thyroid	~0.25 rad

One statement that summarizes some of the information on the effects of radiation doses to the bones has been given. "... visible changes in the skeleton have been reported only after hundreds of rep were accumulated and tumors only after 1500 or more."[37] Another statement that summarizes some of the information on short-term effects of doses to the gastrointestinal tract states: "... though the gastrointestinal tract is one of the sensitive systems to ionizing radiation, it also has a most remarkable regenerative and reparative capacity. It takes doses of well over 1000 r to damage the gut permanently in most mammals studied, and it is capable of rapid dramatic recovery of anatomical and functional integrity with doses in the lethal range."[38]

Some evaluation of radiation doses to the thyroid has been given above.

Acknowledgments—Many of the monitoring data reported here have been collected by the Off-Site Radiological Safety group of the Test Organization. This group has been operated since 1955 by the U.S. Public Health Service, in cooperation with the U.S. Atomic Energy Commission, under the leadership of Mr. Oliver Placak, (USPHS). Additional data originally collected by Mr. Kermit Larson of the University of California at Los Angeles for research purposes, have been very useful as documentary evidence.

During the spring and autumn 1951 test series the off-site monitoring activities were conducted by the H-Division of the Los Alamos Scientific Laboratory, and in the spring of 1952 and spring of 1953 by military personnel augmented by representatives from the U.S. Public Health Service and the Los Alamos Scientific Laboratory. During the test series in the autumn 1951, spring and autumn 1952, and spring 1953, the Health and Safety Laboratory of the New York Operations Office, AEC, operated mobile teams out to 300 miles from the Nevada Test Site, as well as a network of automatic γ-recording stations. The recording stations also were operated by them during the series in the spring 1955.

REFERENCES

1. *The Nature of Radioactive Fallout and its Effects on Man*, Part I, pp. 172–175 and 248–250. Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, May 27–29 and June 3, 1957.
2. G. C. Dale and J. C. Bomyer, *Dose Rates from Ground Contaminated with Residual Radioactive Materials from an Atomic Explosion.* Atomic Weapons Research Establishment (United Kingdom), Report No. 0-35/36 (X) (Official Use only) (1956).
3. G. M. Dunning (Editor), *Radioactive Contamination of Certain Areas in the Pacific Ocean From Nuclear Tests.* U.S. Atomic Energy Commission, Superintendent of Documents, Washington, D.C.
4. *Atomic Energy Commission's Twenty-third Semi-Annual Report to Congress* pp. 282–283. Superintendent of Documents, Washington, D.C. (1957).
5. *The Biological Effects of Atomic Radiation (Summary Reports)* p. 29. National Academy of Sciences and National Research Council, Washington, D.C. (1956).
6. *Radiology* **68,** 2, 260–261 (1957).
7. *Atomic Energy Commission's Twenty-third Semi-Annual Report to Congress* pp. 278–281. Superintendent of Documents, Washington, D.C. (1957).
8. *The Nature of Radioactive Fallout and its Effect on Man*, Part I, p. 181. Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, May 27–29 and June 3, 1957.

9. G. M. Dunning, *Nucleonics* **14,** 2, 38–41 (1956).
10. WT-204 *Operation Ranger*, Vol. 5 (Secret) (1952).
11. WT-425 *Operation Buster-Jangle* (Secret) (1953).
12. *Atomic Energy Commission's Thirteenth Semi-Annual Report to Congress*, p. 117. Superintendent of Documents, Washington, D.C. (1952).
13. *Atomic Energy Commission's Fourteenth Semi-Annual Report to Congress* p. 51. Superintendent of Documents, Washington, D.C. (1953).
14. *Atomic Energy Commission's Eighteenth Semi-Annual Report to Congress* p. 81. Superintendent of of Documents, Washington, D.C. (1955).
15. *Atomic Energy Commission's Twenty-third Semi-Annual Report to Congress* p. 28. Superintendent of Documents, Washington, D.C. (1957).
16. W. H. Langham, *Amer. Industr. Hyg. Assoc. Quart.* **17,** 3, 305–318 (1956).
17. K. Z. Morgan, *Maximum Permissible Concentrations of Radon in Air.* Unpublished Memorandum 27 October, 1954.
18. M. Eisenbud and J. Harley, *Science* **117,** 3033, 141–147 (1953).
19. W. F. Bale, *Hazards Associated with Radon and Thoron.* Unpublished Memorandum, 14 March, 1951.
20. J. Shapiro, *An Evaluation of the Pulmonary Radiation Dosage From Radon and Its Daughter Products.* UR-298 (1954).
21. D. A. Holaday *et al.*, *Control of Radon and Daughters in Uranium Mines and Calculations on Biologic Effects.* U.S. Dept. of Health, Education and Welfare, Public Health Service Publication No. 494 (1957).
22. B. Hultquist, *Kgl. Svenska Vetenskapsakad. Handl* **6,** 3 (1956).
23. G. M. Dunning, *The Scientific Monthly* **81,** 6, 265–270 (1955).
24. A. S. Freeberg, G. S. Kurland, D. L. Chamovitz and A. L. Ureles *J. Clin. Endocrinol.* **12,** 86–111 (1952).
25. A. S. Freeberg, G. S. Kurland and H. L. Blumgart, *J. Clin. Endocrinol. Metab.* **12,** 1315–1348 (1952).
26. D. E. Clark, *J. Amer. Med. Assoc.* **159,** 1007–1009 (1955).
27. *Pathologic Effects of Atomic Radiation* pp. II–8 National Academy of Sciences and National Research Council, Washington, D.C. (1956).
28. G. M. Dunning, Criteria for Establishing Short Term Permissible Ingestion of Fallout Material. *Amer. Industr. Hyg. Quart.* **19,** 2,111–120 (1958).
29. H. F. Hunter and N. E. Ballou, *Simultaneous Slow Neutron Fission of* U^{235} *Atoms. I. Individual and Total Rates of Decay of the Fission Products.* U.S. Naval Radiological Defence Laboratory, San Francisco 24, Calif. (1949).
30. R. C. Bolles and N. E. Ballou, *Calculated Activities and Abundances of* U^{235} *Fission Products.* U.S. Naval Radiological Defense Laboratory, San Francisco 24, Calif. (1956).
31. WT-204 *Operation Ranger*, Vol. 5 (Secret) (1952).
32. WT-425 *Operation Buster-Jangle* (Secret) (1953).
33. *Atomic Energy Commission's Thirteenth Semi-Annual Report to Congress* p. 117. Superintendent of Documents, Washington, D.C. (1952).
34. *Atomic Energy Commission's Fourteenth Semi-Annual Report to Congress* p. 51. Superintendent of Documents, Washington, D.C. (1953).
35. *Atomic Energy Commission's Eighteenth Semi-Annual Report to Congress* p. 81. Superintendent of Documents, Washington, D.C. (1955).
36. *Atomic Energy Commission's Twenty-third Semi-Annual Report to Congress.* Superintendent of Documents, Washington, D.C.(1957).
37. *The Biological Effects of Atomic Radiation* (*Summary Reports*), p. 39. National Academy of Sciences and National Research Council, Washington, D.C. (1956).
38. R. Conard, *Some Effects of Ionizing Radiation on the Physiology of the Gastrointestinal Tract—A Review.* Lecture and Review Series No. 56-2. Naval Medical Research Institute, Bethesda, Maryland (1956).

Health Physics Pergamon Press 1958. Vol. 1. pp. 268–275. Printed in Northern Ireland

THE NATURALLY OCCURRING α-RAY ACTIVITY OF FOODS

R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD
Physics Department, Institute of Cancer Research, Royal Cancer Hospital, London, S.W.3

(*Received* 31 *July* 1958)

Abstract—Most of the activity is due to members of the thorium and radium series. There is a factor of 20,000 between the most active and least active foods. The most active food so far measured is the Brazil nut; breakfast cereals have the next highest α-activity. In general, values in milk products, fruits and vegetables are low. The α-activity seems to increase with the phosphorus content of foods. In general, an adequate diet will not contain less than 2×10^{-12} c of α/day.

TECHNIQUE OF MEASUREMENT

USING the simple technique described elsewhere[1,2] measurements have been made of the α-ray activity of a number of human foods. In all cases observations have been made on the ash, and the rise of activity after ashing observed at intervals over a month:

(a) Immediately after ashing.
(b) After 24 or 48 hr.
(c) At intervals of 2 or 3 days.
(d) A final measurement after 28 days.

The times of counting range up to a maximum of about 64 hr, 24 hr being most common. In order to make estimates of the ratio of radiothorium to radium[1,2] at least 1500 counts have been regarded as necessary. If the activity was so low that this number could not be obtained in, say, 64 hr, the ratio has not been given in Tables 3 and 4. At even lower activities, of the order of 2–3 counts/hr, the activity has been quoted as being less than 10^{-13} c/100 g of starting material.

EXPERIMENTAL RESULTS

In general, the activity is due to members of either the thorium or radium series, but in five instances, namely three teas and two cheeses, the build-up of activity with time was less than expected and, apart from the thorium content, the remaining nuclides of the material could not be adequately identified. It is noteworthy that the ash weights obtained by us for the foods were in good agreement with those quoted by HAWK[3] as shown in Table 1.

Table 1. *Ash weights: A comparison of food weights as obtained by* HAWK *et al., and by us*

Food	HAWK *et al.* (%)	Expt (%)
Brazil nuts	3.4	3.4
Soya flour	4.6	4.6
Mushrooms	1.1	1.1
Milk (evaporated)	1.5	1.7
Brussels sprouts	1.3	1.0
Bananas	0.8	0.9
Butter	2.5	2.6
Bacon	4.3	4.6

For certain foods the activity is so high that the value for the raw material has also been observed directly. These results are in good agreement with those derived from ash measurement if due allowance be made for the change of concentration and effective atomic number during ashing. Examples of active foods for which both the ash and the raw material have been measured are given in Table 2. The values of total α-activity, radiothorium-228 content and radium-226 content for a number of foods are given in Tables 3 and 4. Table 3

Table 2. α-Activity in common foods: A comparison of measurements made on the raw food and on the ash

Food	Total activity observed		Ash concentrations	Total α-activity per g raw material derived from ash value (curies)
	Per g raw material ($Z^{2/3} \sim 4$) (curies)	Per g ash ($Z^{2/3} = 5.1$) (curies)		
Indian tea	3.5×10^{-13}	1.6×10^{-12}	4.2	3.9×10^{-13}
Soya flour	2.4×10^{-14}	5.7×10^{-13}	21.8	2.6×10^{-14}
Shredded wheat	5.5×10^{-13}	5.6×10^{-12}	9.7	5.8×10^{-13}
Brazil nut	3.3×10^{-12}	8.4×10^{-11}	26.6	3.1×10^{-12}

shows the foods in descending order of activity, and Table 4 the same values but arranged in different food categories.

The first and most obvious comment on these tables is the tremendous range of activity encountered, a factor of at least 20,000 being observed so far between the most active and least active foods. In most cases only one specimen of each material has been measured, the specimens being "grab" samples obtained by normal purchase at local food stores. In certain instances of particular interest numbers of samples have been measured and the range of values recorded. It is evident from Table 4 that this range, within a single category of food, is lower by orders of magnitude than that over the whole series of foods. The most active food so far encountered is the Brazil nut, which is well known as having a high content of other trace metals such as barium.[4] Apart from peanuts, other nuts so far examined have much lower activities.

It should however be realized that the values given in Tables 3 and 4 may not necessarily be representative, for we have no evidence so far about the activities of the same food grown in different parts of the world on varying soils. Further studies of this point are being made.

Breakfast cereals comprise the next highest group of activities, the values for materials purchased in the United Kingdom being, in general, higher than their American equivalents though in one notable instance the reverse is true. The reason for these differences is at present quite unknown.

Many categories of foods show relatively little activity, for example milks and milk products, fruits and vegetables, and most meats and some fish. The higher value for sardines arises possibly from the presence of bone. It is interesting to note that the categories of food fall into order in much the same way as when classified according to trace metal content.[5,6]

The list as a whole shows that there is no correlation of radioactivity with the calcium content of the food, but it is interesting to note that there seems to be some relation between the phosphorus content[3,7] and the activities observed by us (Fig. 1).

From Tables 3 and 4 it is possible to make an estimate of total ingested activity for a number of different diets. An adequate diet will not, in general, contain less than $1\text{–}2 \times 10^{-12}$ c of α-activity per day or $2\text{–}3 \times 10^{-13}$ g Ra per day. This figure is lower than suggested by Muth *et al.*[8,9] from a study of the radium content of a much smaller number of foods. However, the inclusion of even small quantities of the most active foods may easily raise the intake by factors of many hundreds. For example, the ingestion of 50 g of Brazil nuts as a single portion can put into the body 1.4×10^{-10} g Ra, namely 700 times the daily intake from the diet quoted above. It will also be noted that this activity is approximately equal to the whole body burden thought to be characteristic of normal individuals.[1,9,10,11]

Our results must still be regarded as preliminary but emphasize the need for investigation into such subjects as the relative activity of similar foods grown in different parts of the world, and the possible differences in body

Table 3. Total α-activity, radium-226 activity and radiothorium-228 activity in foods (arranged in order of decreasing magnitude)

Foodstuff	No. of samples	Total α-activity/ 100 g × 10^{-12} c		Ra^{226}/100 g × 10^{-12} c	Rad-Th^{228}/ 100g × 10^{-12} c	Rad-Th/Ra
		Max	Min	Max	Max.	
Brazil nuts	4	1690	445	273	120	0.4
Buttered brazils	1	290	—	52	17	0.3
Brazil nut chocolate	1	133	—	24	7	0.3
Marzipan brazils	1	105	—	14	10	0.7
Cereal 1 (UK)	1	58	—	6.8	6.1	0.9
Cereal 2 (UK)	3	48	12	6.2	4.6	0.7
Tea (China)	1	46	—	5.1	5.1	1.0
Tea (Indian)	1	39	—	4.8	4.0	0.8
Tea (Paraguayan)	1	37	—	4.3	4.0	0.9
Tea (Brazilian)	1	34	—	4.0	3.6	0.9
Cockles	2	31	11	—	—	Ra and Th series
Cereal 3 (UK)	1	21	—	2.5	2.2	0.9
Flours	5	14.7	0.9	—	—	Variable
Peanuts	1	12.0	—	1.8	0.9	0.5
Peanut butter	1	12.0	—	0.9	1.7	2.0
Mussels	1	10.0	—	—	—	Ra and Th series
Chocolates (mixed)	4	8.0	1.4	—	—	Ra and Th series
Tapioca	1	7.6	—	—	—	Ra and Th series
Drinking chocolate	1	6.7	—	—	—	Ra and Th series
Molasses	1	5.4	—	—	—	Ra and Th series
Beverage 1	1	5.3	—	—	—	Ra and Th series
Cereal 4 (UK)	1	5.2	—	—	—	Ra and Th series
Cereal 5 (USA)	1	4.9	—	—	—	Ra and Th series
Curry powder	1	4.8	—	—	—	Ra and Th series
Cereal 6	1	4.6	—	—	—	Ra and Th series
Prawns (peeled)	1	4.4	—	—	—	—
Vichy water	1	4.0	—	—	—	No Th series
Cereal 7 (USA)	1	4.0	—	—	—	Ra and Th series
Cereal 8 (USA)	1	3.8	—	—	—	Ra and Th series
Sardines (tinned)	1	3.7	—	0.60	0.24	0.4
Cereal 9 (UK)	1	3.1	—	—	—	Ra and Th series
Breads (UK)	3	3.1	0.8	—	—	Ra and Th series
Cereal 10 (USA)	1	3.0	—	—	—	—
Cereal 11 (UK)	1	2.9		—	—	—
Walnuts	1	2.9	—	0.30	0.30	1.0
Soya	1	2.6	—	—	0.52	Th. series only
Cereal 12 (UK)	1	2.4	—	—	—	Ra and Th series
Cereal 13 (UK)	1	2.4	—	—	0.50	Th series only
Haricot beans	1	2.3	—	—	—	—
Biscuits	3	2.2	0.4	—	—	Variable
Baby rusks (UK)	1	2.0	—	—	—	Ra and Th series
Strawberry jam	1	1.9	—	0.24	0.19	0.8

Table 3—continued

Foodstuff	No. of samples	Total α-activity/ 100 g × 10^{-12} c		Ra^{226}/100 g × 10^{-12} c	Rad-Th^{228}/ 100 g × 10^{-12} c	Rad-Th/Ra
		Max	Min	Max	Max	
Salami	1	1.9	—	0.20	0.20	1.0
Cereal 14 (UK)	1	1.9	—	—	—	Ra and Th series
Almonds	1	1.9	—	—	—	Ra and Th series
Calf liver	1	1.8	—	0.10	0.25	2.5
Cashew nuts	1	1.8	—	—	—	—
Salmon (tinned)	1	1.8	—	—	—	—
Sausages	2	1.6	1.1	0.20	0.17	0.9
Betel nuts	1	1.6	—	—	—	Ra and Th series
Cheese 1	1	1.5	—	—	—	Ra and Th series
Cereal 15 (UK)	1	1.4	—	—	—	—
Cereal 16 (USA)	1	1.3	—	—	—	Ra and Th series
Cheese 2	1	1.3	—	0.15	0.15	1.0
Margarine	1	1.3	—	—	—	Ra and Th series
Plaice	1	1.3	—	0.15	0.15	1.0
Cereal 17 (UK)	1	1.2	—	—	—	Ra and Th series
Hazel nuts	1	1.2	—	—	—	—
Evaporated milks (tinned)	2	1.2	0.7	—	—	Ra and Th series
Cereal 18 (UK)	1	1.1	—	—	—	Ra and Th series
Brazil coffee	1	1.1	—	—	—	Ra and Th series
Coconut	1	1.1	—	—	—	—
Fruit salad (tinned)	1	1.1	—	—	—	—
Cereal 19 (USA)	1	0.9	—	—	—	—
Processed cheese	1	0.9	—	—	—	Ra and Th series
Cheese 3	1	0.9	—	—	—	Ra and Th series
Parsnips	1	0.8	—	—	—	—
Pears (tinned)	1	0.7	—	0.11	0.06	0.5
Calf kidney	1	0.7	—	—	—	Ra and Th series
Bacon	1	0.7	—	—	0.14	Th series only
Ice cream	1	0.7	—	—	—	Ra and Th series
Veal	1	0.7	—	0.09	0.06	0.6
Egg	1	0.7	—	0.20	—	Ra series only
Dried currants	1	0.7	—	—	—	Ra and Th series
Butter	1	0.6	—	—	—	—
Beef	1	0.5	—	—	—	Ra and Th series
Pistachio nuts	1	0.5	—	—	—	Ra and Th series
Cod	1	0.4	—	—	—	—
Tomato soup (tinned)	1	0.4	—	—	0.08	Th series only
Peaches (tinned)	1	0.4	—	—	—	Ra and Th series
Pearl barley	1	0.4	—	—	—	—
Blackcurrant juice	1	0.3	—	—	—	—
Sago	1	0.3	—	—	—	Ra and Th series
Cow milks	20	0.3	0.08	—	—	Variable

Table 3—continued

Foodstuff	No. of samples	Total α-activity/ 100 g × 10^{-12} c		Ra^{226}/100 g × 10^{-12} c	Rad-Th^{226}/ 100 g × 10^{-12} c	Rad-Th/Ra
		Max	Min	Max	Min	
Potatoes	1	0.2	—	—	—	—
Peas (tinned)	1	0.2	—	—	—	—
Lamb	1	0.1	—	—	—	Ra and Th series
Corned beef	1	<0.1				
Whale meat	1	<0.1				
Cream	1	<0.1				
Carrots	1	<0.1				
Sprouts	1	<0.1				
Mushrooms	1	<0.1				
Custard powders	2	<0.1				
Fruit jelly	1	<0.1				
Macaroni	1	<0.1				
Macaroni	1	<0.1				
Pineapple (tinned)	1	<0.1				
Banana	1	<0.1				
Spaghetti (tinned)	1	<0.1				
Apple	2	<0.1				
Tomato	1	<0.1				
Sugar	2	<0.1				
Rice	2	<0.1				
Cornflour	1	<0.1				
Beverage 2	1	<0.1				
Baked beans (tinned)	1	<0.1				
Beverage 3	1	<0.1				
Beverage 4	1	<0.1				

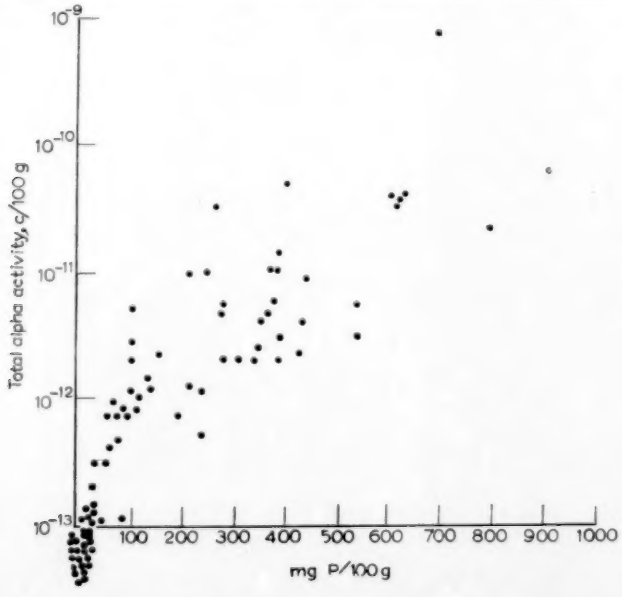


FIG. 1. α-Activity of foodstuffs as a function of phosphorus content.

Table 4. Total α-Activity, radium-226 activity and radiothorium-228 activity (arranged according to classes of food)

	Total α-activity/ 100 g × 10^{-12} c	Ra^{226}/100 g × 10^{-12} c	Rad-Th^{228}/ 100 g × 10^{-12} c	Rad-Th/Ra
Breakfast cereals				
Cereal 1 (UK)	58.0	6.8	6.1	0.9
Cereal 2 (UK)	48.0	6.2	4.6	0.7
Cereal 3 (UK)	21.1	2.5	2.2	0.9
Cereal 4 (UK)	5.2	—	—	Ra and Rad-Th
Cereal 5 (USA)	4.9	—	—	Ra and Rad-Th
Cereal 6 (UK)	4.6	—	—	Ra and Rad-Th
Cereal 7 (USA)	4.0	—	—	Ra and Rad-Th
Cereal 8 (USA)	3.8	—	—	Ra and Rad-Th
Cereal 9 (UK)	3.1	—	—	Ra and Rad-Th
Cereal 10 (USA)	3.0	—	—	—
Cereal 11 (UK)	2.9	—	—	—
Cereal 12 (UK)	2.4	—	—	Ra and Rad-Th
Cereal 13 (UK)	2.4	—	0.50	Th series only
Rusks (UK)	2.0	—	—	Ra and Rad-Th
Cereal 14 (UK)	1.9	—	—	Ra and Rad-Th
Cereal 15 (UK)	1.4	—	—	—
Cereal 16 (USA)	1.3	—	—	Ra and Rad-Th
Cereal 17 (UK)	1.2	—	—	Ra and Rad-Th
Cereal 18 (UK)	1.1	—	—	Ra and Rad-Th
Cereal 19 (USA)	0.9	—	—	—
Nuts				
Brazils	1690	273	120	0.4
Peanuts	12.0	1.8	0.9	0.5
Peanut butter	12.0	0.9	1.7	~2.0
Walnuts	2.9	0.3	0.3	1.0
Almonds	1.9	—	—	Ra and Rad-Th
Cashews	1.8	—	—	—
Betels	1.6	—	—	Ra and Rad-Th
Hazels	1.2	—	—	—
Coconut	1.1	—	—	—
Pistachios	0.5	—	—	Ra and Rad-Th
Confectionery				
Buttered brazils	290	52	17	0.3
Brazil nut chocolate	133	24	7	0.3
Marzipan brazils	105	14	10	0.7
Chocolates (mixed)	8.0	—	—	Ra and Rad-Th
Molasses	5.4	—	—	Ra and Rad-Th
Biscuits (mixed)	2.2	—	—	Variable
Strawberry jam	1.9	0.24	0.19	0.8
Ice cream	0.7	—	—	Ra and Rad-Th
Sugar	<0.1	—	—	—
Custard powder	<0.1	—	—	—

Table 4—continued

	Total α-activity/ 100 g × 10^{-12} c	Ra^{226}/100 g × 10^{-12} c	Rad-Th^{228}/ 100 g × 10^{-12} c	Rad-Th/Ra
Beverages				
Tea (China)	51	5.7	5.7	1.0
Tea (Indian)	39	4.8	4.0	0.8
Tea (Paraguayan)	37	4.3	4.0	0.9
Tea (Brazilian)	34	4.0	3.6	0.9
Drinking chocolate	6.7	—	—	Ra and Rad-Th
Meat Extract 1	5.3	—	—	Ra and Rad-Th
Vichy water	4.0	—	—	—
Brazil coffee	1.1	—	—	Ra and Rad-Th
Tomato soup (tinned)	0.4	—	0.08	Th series only
Meat extract 2	<0.1	—	—	—
Beverage 1	<0.1	—	—	—
Beverage 2	<0.1	—	—	—
Meats and fish				
Cockles	31	5.7	1.9	0.3
Mussels	10.0	1.82	0.61	0.3
Sardines (tinned)	3.7	0.60	0.24	0.4
Salami	1.9	0.20	0.20	1.0
Calf liver	1.8	0.10	0.25	2.5
Salmon (tinned)	1.8	—	—	—
Sausages	1.6	0.20	0.17	0.9
Plaice	1.3	0.15	0.15	1.0
Calf kidney	0.7	—	—	Ra and Rad-Th
Bacon	0.7	—	0.14	Th series only
Veal	0.7	0.09	0.06	0.60
Egg	0.7	0.20	—	Ra series only
Beef	0.5	—	—	Ra and Rad-Th
Cod	0.4	—	—	×
Lamb	0.1	—	—	Ra and Rad-Th
Corned beef	<0.1	—	—	—
Whale meat	<0.1	—	—	—
Fruit and vegetables				
Haricot beans	2.3	—	—	—
Fruit salad (tinned)	1.1	—	—	—
Parsnips	0.8	—	—	—
Pears (tinned)	0.7	0.11	0.06	~0.5
Dried currants	0.7	—	—	Ra and Rad-Th
Peaches (tinned)	0.4	—	—	Ra and Rad-Th
Blackcurrant juice	0.3	—	—	—
Potatoes	0.2	—	—	—
Peas (tinned)	0.2	—	—	—
Carrots	<0.1	—	—	—
Sprouts	<0.1	—	—	—
Fruit jelly	<0.1	—	—	—

Table 4—continued

	Total α-activity/ 100 g × 10^{-12} c	Ra^{226}/100 g × 10^{-12} c	Rad-Th^{228}/ 10^{-12} c	Rad-Th/Ra
Fruit and Vegetables (cont.)				
Pineapple (tinned)	<0.1	—	—	—
Banana	<0.1	—	—	—
Apples	<0.1	—	—	—
Tomatoes	<0.1	—	—	—
Baked beans (tinned)	<0.1	—	—	—
Mushrooms	<0.1	—	—	—
Milks and milk products				
Cheese 1	1.5	—	—	Ra and Rad-Th
Cheese 2	1.3	0.15	0.15	1.0
Evaporated milks (tinned)	1.2	—	—	Ra and Rad-Th
Margarine	1.2	—	—	—
Processed cream cheese 3	0.9	—	—	Ra and Rad-Th
Cheese 4	0.9	—	—	Ra and Rad-Th
Butter	0.6	—	—	—
Cow milks	0.2	—	—	Variable
Full cream	<0.1	—	—	—
Miscellaneous cereal foods				
Tapioca	7.6	—	—	Ra and Rad-Th
Soya	2.6	—	0.52	Th series only.
Pearl barley	0.4	—	—	—
Sago	0.3	—	—	Ra and Rad-Th
Macaroni	<0.1	—	—	—
Spaghetti	<0.1	—	—	—
Rice	<0.1	—	—	—
Cornflour	<0.1	—	—	—
Bread and flours	14.7–0.8	—	—	Variable

burden which might be expected of populations living on different diets.

It should be stated that up to the present there is no evidence that the above levels of radioactivity in foods have any biological consequence.

Acknowledgment—We are indebted to Dr. R. HASTERLIK of the Argonne Cancer Research Hospital for the specimens of American cereals examined by us.

REFERENCES

1. R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD, *Nature, Lond.* **181,** 518 (1958).
2. R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD, *Brit. J. Rad.* In press.
3. P. B. HAWK, B. L. OSER and W. H. SUMMERSON, *Practical Physiological Chemistry.* Churchill, London (1954).
4. W. SEABER, *The Analyst* **58,** 575 (1933).
5. E. J. UNDERWOOD, *Trace Elements in Human and Animal Nutrition.* Academic Press, New York (1956).
6. G. W. MONIER-WILLIAMS, *Trace Elements in Food.* Wiley, New York (1950).
7. R. A. MCCANCE and E. M. WIDDOWSON, *The Chemical Composition of Foods.* Medical Research Council Special Report Series No. 235. H.M.S.O. London (1946) (Reprinted 1956).
8. H. MUTH, A. SCHRAUB K. AURAND and H. J. HANTKE, *Brit. J. Rad.* Suppl. No. 7, p. 54. British Institute of Radiology, London (1957).
9. H. MUTH, B. RAJEWSKY H. J. HANTKE and K. AURAND, *Strahlentherapie* **104,** 2 (1957).
10. J. B. HURSH *Brit. J. Rad.* Suppl. No. 7, p. 45. British Institute of Radiology, London (1957).
11. F. W. SPIERS and R. J. BURCH, *Brit. J. Rad.* Suppl. No. 7, p. 81. British Institute of Radiology, London (1957).

Health Physics Pergamon Press 1958. Vol. 1. pp. 276–281. Printed in Northern Ireland

APPLICATIONS OF AN ANALOG COMPUTER TO ANALYSIS OF DISTRIBUTION AND EXCRETION DATA*

B. R. FISH
Health Physics Division, Oak Ridge National Laboratory,†
Oak Ridge, Tennessee

(*Received* 11 *June* 1958)

Abstract—The work of the ORNL Health Physics Division in studying various mathematical models for the distribution and excretion of uranium following intravenous injection has been greatly facilitated by the use of an electronic analog computer. A four compartment linear model appeared to be adequate to explain data obtained from male rats and low-dose level human data. However, to simulate human excretion following injection at higher dose levels, it was necessary to superimpose a saturable excretion pathway upon the linear model. A prime object of the model work has been the derivation of some means whereby field measurements of excretion may be related to internal dose. That object has been realized in the inverse model by comparing the model output with the excretion data to obtain an error function which is used to drive the model.

INTRODUCTION

It is often necessary for the health physicist to design experiments or equipment, to analyze data, or to demonstrate the action of a device. The ease with which one may accomplish those functions through the use of an electronic analog computer has shown that instrument to be a very powerful tool.

There are many types of analog computers. They might be characterized as mechanical, optical, hydraulic, electronic or as is usually the case, a combination of these forms. Examples of mechanical analog computers include the slide rule and the automobile speedometer. The count rate circuits of radiation survey meters are fine examples of electronic analog computers. Since the electronic form appears to be by far the most flexible, these remarks shall be confined to that particular category of analog devices.

Fig. 1 shows schematically some of the basic circuit elements of which an electronic analog computer may be constructed. Voltages are commonly used to represent the variables. In addition to the units shown here there are many circuits which may be used for nonlinear operations such as limiting, sticking, dead space and transport lag.

In the case of many commercially available computers the problem may be wired on a removable patch panel before it is placed on the machine. The necessary computer components are interconnected on the panel by means of special plugs and jumper cords. Any desired change in the wiring may be made as the need becomes apparent.

As for the question of accuracy of an analog computer, one must remember that the components do not have ideal characteristics. In general, it is difficult to give a quantitative estimate of accuracy before running a problem that is at all complicated. However, practical experience has shown that when a problem is set up with due consideration being given to the choice of alternate means of arranging the various operations, it is usually possible to obtain solutions which deviate from the true solutions anywhere from 0.1 to about 10 per cent, depending upon the problem. Detailed means for evaluating the errors involved in a

* Revision of a paper presented at the Health Physics Society Meeting, University of Pittsburgh, Pennsylvania, 17–19 June, 1957.

† Operated for the U.S. Atomic Energy Commission by Union Carbide Corporation.

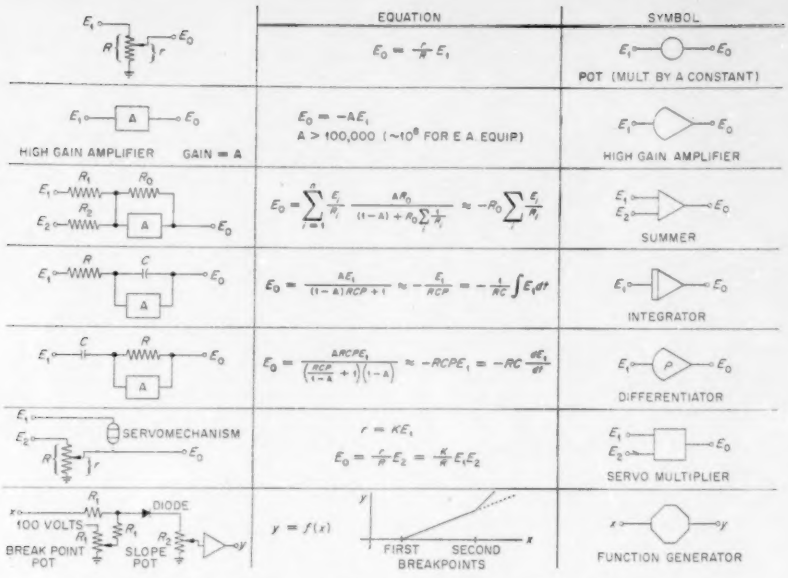


FIG. 1. Basic components of an electronic analog computer.

particular problem are outlined in the general references below. One should first learn where to expect the large sources of error to occur and try to avoid them, of course, but keeping in mind that ultra-accuracy is not the primary objective in the use of analog computers.

Four compartment linear system

The work on a model for the distribution and excretion of hexavalent uranium will serve to illustrate how an analog computer may be applied to a practical problem. Fig. 2 shows the four compartment model that has been discussed in detail by BERNARD.[1] Briefly, an initial injection is made into the blood compartment. The amount, S_1, in the blood is given up at a rate of $\lambda_1 S_1$. A fraction, f, goes to the bone. Of the amount that does not go to the bone the fraction, ϕ_1, is held up in the kidney, and the remainder is transferred directly from the blood to urine via kidney with negligible delay. The performance equations are linear and are not difficult to solve analytically. However, a great deal of speed and flexibility are obtained by simulating the system on the analog computer. For example, this version of the model may be set up on the computer in ten minutes and separate runs need take only 3–5 min.

The S_1 integrator (Fig. 2) starts out with positive 100 V which represents 100 per cent of the injected dose. As the run progresses the output of the first integrator is $-S_1$ and the output of the λ_1 pot is $-\lambda_1 S_1$ which is added along with $+\lambda_2 S_2$ into the S_1 integrator input so that their sum is the time rate of change of S_1 as the first equation prescribes. Of the quantity, $-\lambda_1 S_1$, a fraction, f, goes to the bone.

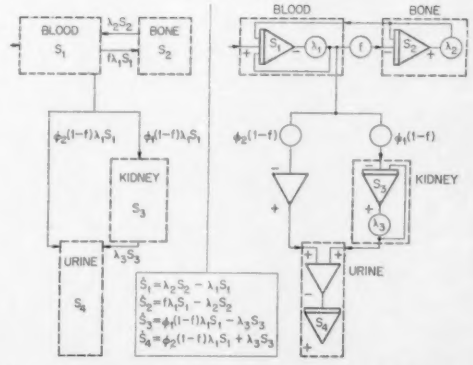

BLOOD
BONE
KIDNEY
URINE
$\dot{S}_1 = \lambda_2 S_2 - \lambda_1 S_1$
$\dot{S}_2 = f\lambda_1 S_1 - \lambda_2 S_2$
$\dot{S}_3 = \phi_1(1-f)\lambda_1 S_1 - \lambda_3 S_3$
$\dot{S}_4 = \phi_2(1-f)\lambda_1 S_1 + \lambda_3 S_3$

FIG. 2. Three compartment model compared with the related computer analog.

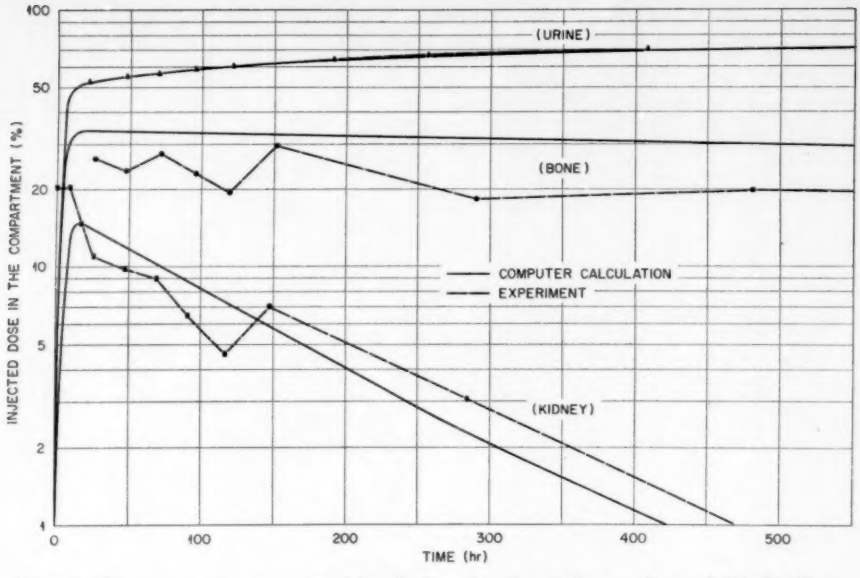


FIG. 3. Three compartment model solution for the male rat data of W. F. NEUMAN.

A similar description applies to the dynamic action of the bone and kidney compartments. If, as is commonly the case, the natural time constant of the basic integrator is 1 sec, setting the λ-pots to the numerical values of the decay constants in terms of hours produces a time scale factor such that one second machine time is equivalent to 1 hr real time. To obtain the dynamic response of any compartment in terms of percent of the injected dose it is only necessary to record the output voltage of that compartment.

Fig. 3 shows the results of applying the four compartment model to the data obtained by NEUMAN[2] in a study of the distribution and excretion of uranium in rats. A time scale relating 1 sec on the machine to 5 hr real time was used, such that the data for Fig. 3 required less than 2 min operating time. It should be emphasized that the decay constants and the distribution percentages used in the computer setup were derived from the urine curve alone. The solid curves represent the compartment contents predicted by the computer results.

Table 1 will serve to compare the parameters, for the four compartment model, derived from the rat data with those obtained from the excretion results of a study made by BASSETT[3] at the University of Rochester using low dose level intravenous injections of hexavalent uranium in six human subjects. It may be seen that except for a longer kidney half-life the rat parameters are little different from those of the human subjects.

Table 1. *Calculated values of the model parameters*

	λ_1 (hr^{-1})	λ_2 (hr^{-1})	λ_3 (hr^{-1})	$-f$	$\phi_1(1-f)$	$\phi_2(1-f)$
Rats	0.221	0.00037	0.0045	0.315	0.095	0.590
Rochester patients	0.407	0.00028	0.084	0.225	0.145	0.630

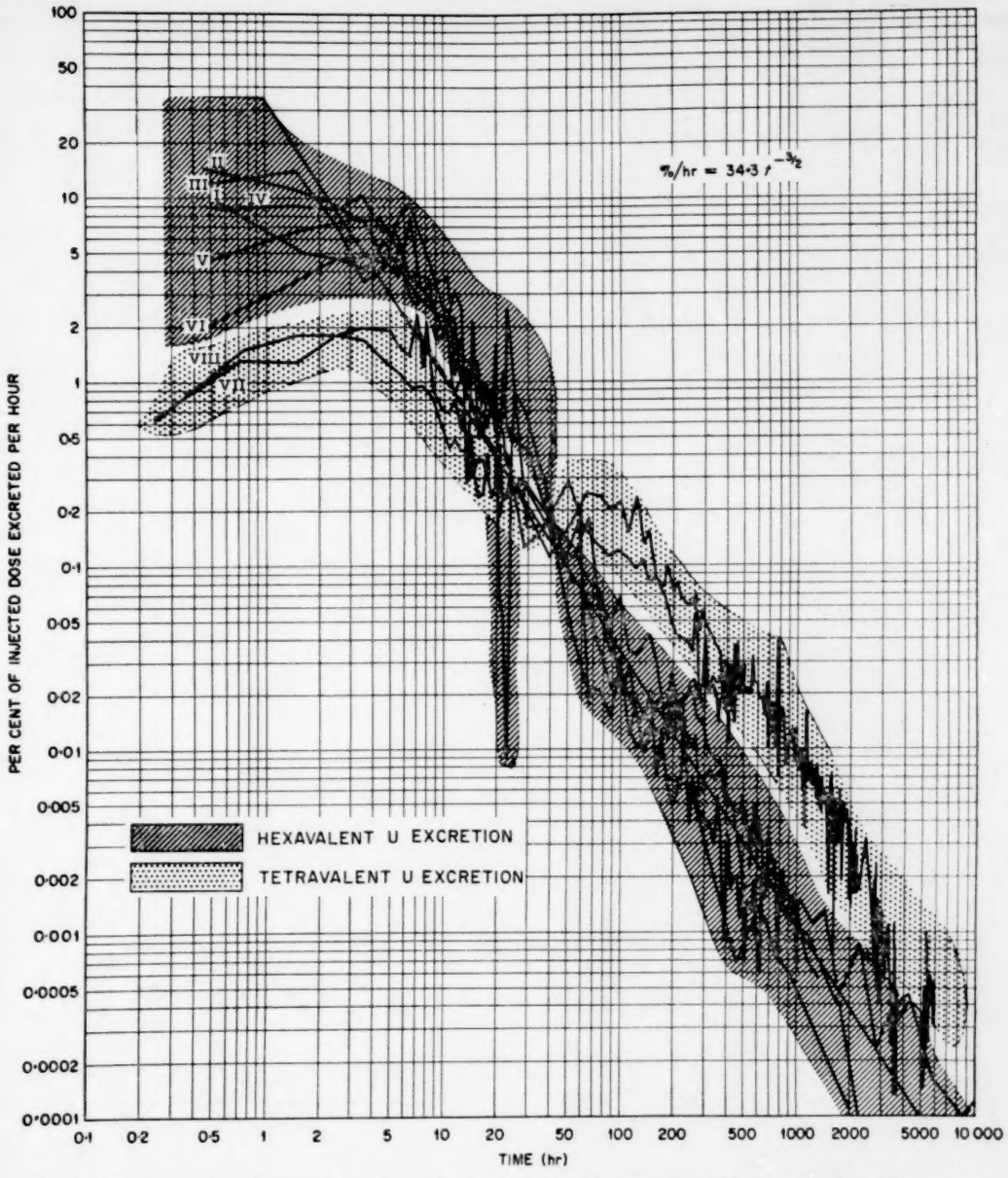


FIG. 4. Urinary rate of excretion for eight terminal brain tumour patients following intravenous injection (S. R. BERNARD and E. G. STRUXNESS).

Gravimetric dose dependence

Fig. 4 shows the excretion results of eight terminal brain tumor patients; the first six were injected intravenously with hexavalent uranium at the Massachusetts General Hospital in Boston (patients VII and VIII received tetravalent uranium which does not follow the four compartment model). The study has been

previously described by BERNARD. It is interesting to note that, although one may approximately fit the excretion data of each patient to a three component exponential function, which is the same form one obtains from the four compartment linear model, the derived decay

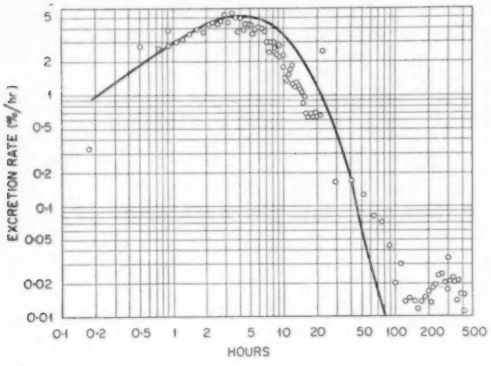


FIG. 5. Saturated model for case of patient BVI.

constants and distribution percentages for the various patients show little relation to each other. However, if one considers the curves in terms of the injected gravimetric dose level then the simple model may be brought back into focus. Patients I, II, and III each received approximately the same initial dose. The initial dose of patient IV was twice the dose received by the first three, while that of patient V was twice as large as patient IV's and, in turn, patient VI received three times the amount given to patient V. This consideration, in which a higher dose results in a lower initial percentage excretion rate, leads directly to the idea that there is some excretion pathway which may become saturated.

On the assumption that the direct path can transport uranyl ions only up to some maximum rate, the model again becomes clear. For two cases that are saturated, i.e. excreting at the same mass rate, the case having the higher initial dose will excrete at a lower percentage rate. The computer is set up to keep the direct path excretion a constant until the amount trying to come through is less than the maximum transfer rate, at which time, relays are used to switch back to the linear setup.

Fig. 5 illustrates the results of applying the saturable model to the excretion data of Boston patient VI, the one receiving the largest injected dose. The significant fact is that the parameters used on the computer setup were not derived from patient VI's data, but were the average values obtained from the Rochester patients, and represent one approach to a general four compartment model for hexavalent uranium. The apparent departure of the data from the model toward the end has been interpreted to be a change in body response just preceeding expiration of the patient. Similar results were obtained for the other Boston patients merely by adjusting one potentiometer which represented the injected dose in terms of micrograms per kilogram.

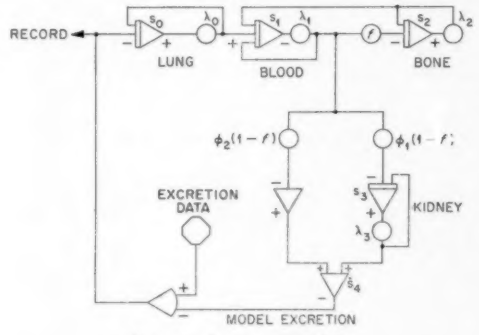


FIG. 6. Inverse model.

Interpretation of excretion data

A major goal of this type of work is represented in Fig. 6. Up to now known inputs have been imposed upon various models and the outputs compared to the measured output data. The question, assuming that one now has an acceptable model, is whether the input function can be derived from a given excretion pattern. The inverse model is an attempt to operate upon urine data to obtain a dynamic estimate of body burden in the several compartments and to infer the form of the input function. Although no satisfactory value of lung half-life is available, a lung compartment is included here to better illustrate the use of the equipment. The operation of the computer setup is rather simple in principle. The set of excretion data is expressed as a function of time by the function generator in which voltage is made to

correspond to uranium excretion rate. If the output of the model is not equal in magnitude and opposite in sign to the excretion data voltage at any particular time, a very large error signal is obtained from the high gain amplifier. The error signal is fed into the lung compartment. If, for example, the model excretion rate were lower in magnitude than the excretion data the difference would be positive, which is changed to negative by the amplifier, and consequently a large negative voltage appears at the input to the lung compartment. A negative input to the lung with this particular setup will increase the model excretion rate to more nearly approximate the excretion data. By the same means if the model excretion rate is too large, the resulting positive error voltage will tend to make it lower. Now, if the input to the model, i.e. the error voltage, is such as to force the accepted model to excrete the same pattern as the given excretion data, then the error function is a measure of the function which describes the rate of uranium input to the lung.

Fortunately the fidelity of this model is easy to test. The procedure is to replace the function generator with another model having the same parameter values as the first. Then, by introducing a known input, say a step function or a triangular wave, into the generating model, a derived input is obtained which may then be compared with the known original. Such tests showed that the inverse model tends to round off any abrupt changes, e.g. rounding the leading edge of a step function, but that the derived inputs are grossly the same as the given functions The departure from a true reproduction of the input function is largely due to the time lag inherent in the change in model output related to a given input change.

A study is currently being performed on the distribution and excretion of uranium in dogs when exposed to multiple inhalations. The computer techniques that have been outlined will be directly applied to aid in the analysis of this and similar work.

Other applications

Other work that might be of interest includes the fallout computer described by WRIGHT[4] and his associates at the National Bureau of Standards. The computer was developed for predicting the geographic pattern of fallout as a function of certain meteorological data and of the initial height of the cloud. Some recent analog work being done in Texas in connection with tracer studies has been described by MACDONALD.[5]

Further applications of analog computers seem to hold much promise. Preliminary investigations have indicated that computation in connection with long-chain decay series, and direct plotting of solutions to non-linear differential equations can be greatly facilitated. Perhaps, the most attractive current application of an analog computer is to extend the model work to include translocation of daughter products between compartments and eventually to the computation of maximum permissible concentration based upon any desired critical organ.

Acknowledgments—The author is deeply indebted to J. R. MUIR for his assistance in performing the work described as well as for his help in preparing this report. Much credit is due also to F. P. GREEN for his aid in obtaining computer time and for his many helpful suggestions on computer setup.

REFERENCES

1. S. R. BERNARD and E. G. STRUXNESS, *A Study of the Distribution and Excretion of Uranium in Man: An Interim Report.* ORNL-2304.
2. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds*, Ch. II, p. 701–727 McGraw-Hill, New York (1949).
3. S. H. BASSETT *et al.*, *The Excretion of Hexavalent Uranium following Intravenous Administration.* II. *Studies on Human Subjects.* University of Rochester, UR–37.
4. J. H. WRIGHT *et al.*, *J. Res. Nat. Bur. Stand.* **58,** No. 2 (1957).
5. J. R. MACDONALD *et al.*, *Rad. Res.* **6,** 585 (1957).

General references

KORN and KORN, *Electronic Analog Computers.* McGraw-Hill (1952).
C. L. JOHNSON, *Analog Computer Techniques.* McGraw-Hill (1956).
W. W. SOROKA, *Analog Methods in Computation and Simulation.* McGraw-Hill (1954).

Health Physics Pergamon Press 1958. Vol. 1. pp. 282–287. Printed in Northern Ireland

BREMSSTRAHLUNG COUNTING OF Sr^{90} INJECTED DOGS*

C. W. MAYS, D. H. TAYSUM, W. FISHER and B. W. GLAD
Radiobiology Laboratory, University of Utah, College of Medicine, Salt Lake City, Utah

(*Received* 25 *June* 1958)

Abstract—Strontium retention has been measured in dogs injected with Sr^{90} by whole body Bremsstrahlung counting. For young adult beagles 17.7 months old at injection, fractional strontium retention may be expressed as $R = 0.68t^{-0.21}$ where t is the number of days after injection. This is slightly less than fractional radium retention measured in beagles of similar injection age at this laboratory. A method has been developed in which low levels of Sr^{90} in living dogs can be determined by Bremsstrahlung counting with a standard deviation of less than 0.1 μc.

INTRODUCTION

STRONTIUM-90 and its daughter, yttrium-90, decay by β-emission. In large mammals, whose skeletons are poisoned with Sr^{90}, most of these β-particles are absorbed by bone and the surrounding tissue. This absorption is accompanied by the emission of Bremsstrahlung X-rays. Hence, although direct measurement of escaping β-particles is impractical in living animals, these Bremsstrahlung X-rays can be γ-counted.

K-9 BARK II, THE IMPROVED DOG COUNTER

Dogs were anesthetized and placed in K-9 Bark II, our whole body gamma detector[1] as shown in Fig. 1. This liquid scintillation counter† was patterned after VAN DILLA's original K-9[2] and the Los Alamos Human Counter.[3] This counter has two very desirable features: (a) high counting efficiency and (b) geometry independence. The high counting efficiency is due to the fact that nearly all radiation leaving the dog is intercepted by the scintillator. The shape of K-9 Bark II and location of the photomultiplier tubes result in counting rates fairly independent of the dog's position and of the radioelement distribution within the body.

STRONTIUM RETENTION

Strontium retention from 10 days to 3 years after injection‡ is plotted in Fig. 2. Each point represents the average fractional retention of a group of from three to six dogs whose injected Sr^{90} doses ranged from 1.7 to 100 μc/kg. Five groups of dogs were measured. Some groups were measured several times. The range of individual values for each group is indicated by bars. Consistent differences observed on repeated measurements showed these individual variations to be real and not due to inaccuracy of measurement. In beagle dogs 17.7 months

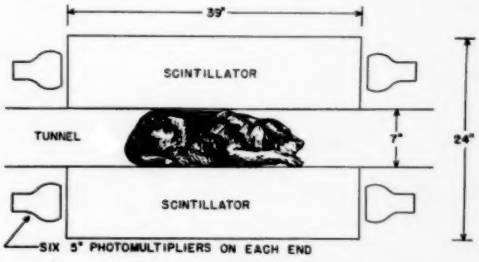


FIG. 1. Dog positioning in K-9 Bark II.

* This work was supported by the U.S. Atomic Energy Commission.

† The scintillator is 70 gal of toluene with 5 g/l. of *p*-terphenyl and 0.2 g/l. of POPOP.

‡ The Sr^{90}, in equilibrium with its daughter, Y^{90}, was given in a single intravenous injection in a sodium citrate, citric acid solution of pH 3.5 and a total citrate concentration of 0.08 M. The dogs were confined separately in metabolism cages, 30 in. by 30 in. by 36 in. from 1 week before injection to 3 weeks after injection. Afterward they were returned to their kennels.

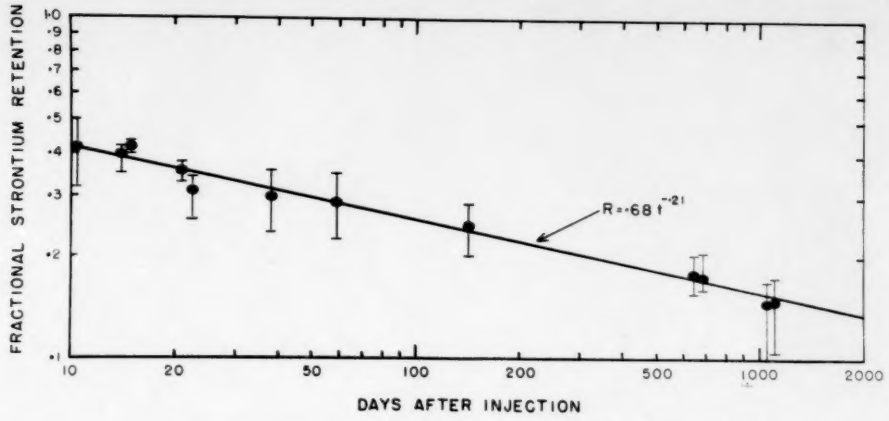


FIG. 2. Fractional strontium retention vs. days after injection.

old* at injection, fractional strontium retention may be expressed by the power function

$$R = 0.68t^{-0.21}$$

where t is the number of days after injection. This equation has been independently verified by the Chemistry Group of the Radiobiology Laboratory from excreta measurements. These results indicate that in our beagles strontium retention is slightly less than radium retention ($R = 0.79t^{-0.20}$).[4] For example, at 1000 days after injection, fractional strontium is 16 per cent, whereas fractional radium retention is 20 per cent.

Strontium retention is expected to decrease as age at injection increases, but at present we do not have enough data to incorporate this effect into our retention equation.

Fractional strontium retention is independent of dose size from 1.7 to 100 μc/kg. That is, dogs receiving high doses of strontium retained the same fraction of their injections as did dogs receiving lower doses. A similar lack of effect of dose size on fractional retention has been observed in our radium dogs[4] and our plutonium dogs.[5]

To test Bremsstrahlung efficiency before skeletal deposition was complete, fourteen dogs were counted within an hour after injection when a large fraction of the injected Sr^{90} and Y^{90} was still in the soft tissue. The average count rate was only 87 per cent of that which would have been obtained if all of the injected Sr^{90} and Y^{90} had deposited in bone. This is because soft tissue, due to its low effective atomic number, is less efficient in Bremsstrahlung production than is bone with its higher effective atomic number.

THE STRONTIUM STANDARD

The best Sr^{90} standard to use for whole body counting is an animal which contains a known amount of strontium in its bones. Thus the efficiency of Bremsstrahlung production and amount of self absorption in the measured animals are duplicated in the standard.

A beagle was injected with 115 μc of Sr^{90}. Five days later, after most of the retained dose was localized in the bone, the dog was sacrificed and frozen. The amount of strontium retained was determined by subtracting the amount excreted from the amount injected. At the end of the experiment, the retained strontium in the standard was directly measured. Analysis of excretion and retention is shown in Table 1.

Activity in the soft tissue was determined by Bremsstrahlung counting, whereas the activity in the excreta and ashed skeleton was determined via β-counting.†

* At this age the long bones are full length (epiphyses closed) but the ribs are still growing. The dogs each weighed about 10 kg.

† The beta counting was done by the Chemistry Group of the Radiobiology Laboratory.

Measured values accounted for about 95 per cent of the injected dose. These values were adjusted by assigning two-thirds of the discrepancy to the excreta and one-third to the retained burden. It is of interest to note that

Table 1. *Excretion and retention by the dog used as the strontium standard*

Sample	Measured % injected dose		Adjusted % Injected dose
Excretion			
Fecal	28.6		
		45.9	49.5
Urinary	17.3		
Retention			
Soft tissue	2.9		
		48.7	50.5
Skeletal	45.8		
Total		94.6	100.0

at 5 days after injection, only 2.9/45.8 = 6 per cent of the retained dose remained in the soft tissue. This shows that strontium, like calcium and radium, concentrates in the skeleton.

TECHNIQUE

To minimize dead time errors, a fast Los Alamos model 503 pulse amplifier was used without an upper gate. This was necessary because some dogs contained large doses, up to 1000 μc,* of Sr^{90}. The gain was adjusted such that a National Bureau of Standards 1 μc Ra^{226} standard gave 1,300,000 counts/min. Each dog was counted for 2 min and background was checked before each count. The strontium standard was run at the start and finish of each series of measurements. When possible, dogs were measured prior to strontium injection to determine their natural γ-count rates. Most of this natural activity is due to potassium-40 and cesium-137. After injection, the dog's natural γ-count rate was subtracted from the total net count rate to give the Bremsstrahlung count rate. This was necessary because whereas over one-half of the K^{40} γ-rays and over one-third of the Cs^{137} γ-rays were counted, only about 1/1500 of the beta disintegrations were detected using this technique. Of Sr^{90} deposited in bone in equilibrium with its daughter, Y^{90}, 1 μc gave about 3000 counts/min. This roughly equalled the average natural γ-count rate from a 10 kg beagle. Hence, the accuracy of the above method (no upper gate, gain such that 1 μc Ra^{226} gives 1,300,000 counts/min) is limited by how accurately the natural γ-count rate can be estimated. An uninjected dog was routinely measured with each group of injected dogs to monitor changes in the natural γ-count rates of the dogs. These changes are mainly due to radioactive fallout contamination in the dog's diet and in their fur. It is advisable to wash animals thoroughly prior to measurement.

Extending the limit of Bremsstrahlung detection

The technique used in the measurement of these strontium poisoned dogs had certain advantages:

(1) The fast circuitry of the Los Alamos model 503 amplifier used without an upper gate minimized dead time errors when counting dogs with large strontium burdens.

(2) Individual natural γ-count rates could be measured prior to injection, and changes in these count rates could be monitored by measuring an uninjected dog with each group of injected dogs.

(3) This method did not require identification of the γ-emitters which were responsible for the dog's natural count rate. This factor would be quite important where only limited equipment is available for γ-ray spectroscopy at very low levels of activity.

(4) Only 2 min of counting time were used for each dog. Yet the total error due to dog positioning, strontium distribution within the skeleton, and statistical fluctuations in count rate ranged between 3 per cent to 10 per cent of the dog's strontium activity.

The chief disadvantage of the preceding method is the limit of detection of Sr^{90} imposed by uncertainty of the natural γ-activity in the body. This limit of sensitivity can be reduced using a different approach.

The trick is in knowing beforehand which radioelements are present in the body. Separate standards of each radioelement are placed in phantoms such that the effects of distribution and self absorption in the living animals are closely approximated. Then the animal and each of these "N" standards are counted under "N" separate conditions. The relative efficiency of detection for each radioelement with respect to the others must be different in each condition. There is no theoretical limit to the number of radioelements which can be resolved by this method. Equations for the simultaneous determination of up to three radioelements in a specimen are derived in the Appendix.

* 100 μc/kg times 10 kg dog weight.

Experimental limit of detection

A 7 kg mongrel with no known radiation exposure* was thoroughly washed with soap and water and sacrificed. A 1 in. diameter plastic tube of $\frac{1}{16}$ in. wall thickness was inserted lengthwise through the dog so that 10 ml ampoules containing aqueous solutions of Sr^{90}—Y^{90} could be placed inside the dog. It was assumed that the principle emitters detected

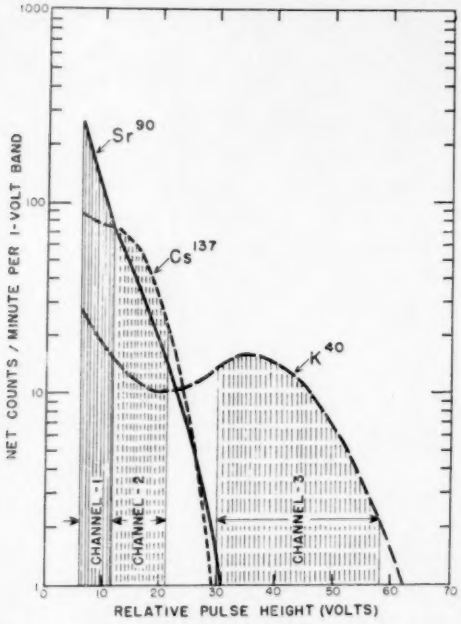


FIG. 3. Spectra of Sr^{90}, Cs^{137} and K^{40} as detected by the whole dog counter, K-9 Bark II. Values correspond to 0.2 μc Sr^{90} deposited in beagle bone and imbedded in a cylinder of paraffin 14 cm in diameter and 18 cm long; 0·0014 μc Cs^{137} centered in a masonite cylinder 15 cm in diameter and 52 cm long; 5 g of potassium centered in wooden cylinder 17 cm in diameter and 24 cm long.

would be Sr^{90}, Cs^{137}, and K^{40}. The dog was counted thirty times in each of three energy channels as shown in Fig. 3. The channels were established so that the efficiency of each emitter relative to the other two would be greatest in a particular channel. The relative efficiency of Sr^{90} was greatest in channel 1, Cs^{137} in channel 2, and K^{40} in channel 3. An additional factor used in establishing channels was to maximize S^2/B, where S was the net counting rate of a given emitter in its chosen channel and B was the background counting rate.

Sr^{90} values were computed for each measurement by the equation for the three radioelement cases which are derived in the Appendix. From the differences between true and computed values of these thirty samples, the standard deviation of a single 4 min measurement was found to be 0.19 μc. If this error were random, then the average of ten measurements (equivalent to a single 40 min measurement) should have a standard deviation of $0.19/\sqrt{10} = 0.06$ μc. As shown in Table 2 the measured deviation of average values corresponds closely to this standard deviation (0.06 μc) indicating that small amounts of Sr^{90} in a dog may be measured by this technique with a standard deviation of less than 0.1 μc.

*Table 2. Accuracy of Sr^{90} detection**

True Sr^{90} content (μc)	Measured Sr^{90} content (μc)	Error
1.0	+0.96	−0.04
0.5	+0.55	+0.05
0.0	−0.12	−0.12

* Each number is the average of ten measurements.

It has been estimated[6] that if Sr^{90} fallout continues at its present rate of 10 mc/mile^2 per year, that the average equilibrium burden of Sr^{90} in adult humans (containing 1 kg of calcium) will reach 0.021 μc. This is not far below our present limit of detection for dogs.

There is no reason why the limit of Bremsstrahlung detection cannot be extended further. Perhaps some reader may be motivated to do this. Some modifications of our whole body counter[1] and our technique which would enhance Bremsstrahlung detection are:

(1) Increase counting stability in the strontium channel. The ratio of observed fluctuations/statistical fluctuations in background counts was 3.1 in the strontium channel as contrasted against 1.4 in the cesium channel and 1.2 in the potassium channel. This was in large part due to photomultiplier tube noise in the low energy (strontium) range.

(2) Reduce the thickness (presently 8.5 in.) of the scintillator. This would reduce background and also decrease the number of interactions of γ-rays from

* The natural γ-activity of this 7 kg dog was found to be equivalent to 14.0 g of potassium and 0.0026 μc of Cs^{137}. The potassium value (0.2 per cent of body weight) is about what we expected, but the cesium value is from two to three times greater in proportion to potassium content than values we have observed in other dogs.

Cs^{137} (0.66 MeV) and K^{40} (1.46 MeV). The weaker Bremsstrahlung X-rays are well absorbed by thin layers.

(3) Increase the area (presently 4 per cent) of K-9 covered by photomultiplier tubes. This would increase the efficiency of light collection from the scintillator. The optimum condition would be to cover all outside walls with photocathode surfaces.

(4) Replace the liquid scintillator with plastic or crystal scintillators giving better scintillation efficiency and better energy resolution. Sr^{90} burdens in dogs ranging from 2.9 to 391 μc have been measured in dogs at the Argonne National Laboratory[7] using a 4 in. diameter, 1.5 in. thick NaI (Tl) crystal. A limit of detection of 0·1 μc was predicted.

(5) Increase the thickness of shielding (presently 3–5 in. of steel) to reduce background counts.

As body burdens of Sr^{90} increase due to atomic fallout, and as techniques improve, it *may* be possible in the not too distant future to detect Sr^{90} in the general population by Bremsstrahlung counting.

SUMMARY

(1) Strontium retention in dogs is slightly less than radium retention.

(2) The size of the injected strontium dose from 1.7 to 100 μc/kg has no effect on fractional strontium retention.

(3) Low levels of Sr^{90} in living dogs can be measured with a standard deviation of less than 0.1 μc.

Acknowledgments—The pioneering Bremsstrahlung measurements by M. A. Van Dilla[8,9] in the old K-9[2] served as a valuable guide for this work. Special thanks is also due to B. J. Stover and D. R. Atherton of the Chemistry Group at the Radiobiology Laboratory for preparation of injection solutions and for independently checking our retention equation by excreta measurements.

APPENDIX

Simultaneous determination of N *radioelements*

If a specimen is known to contain several different radioelements, their individual activities may be simultaneously determined when the radiation from each radioelement can be detected and the comparative efficiencies of detection among the radioelements are different under various conditions. It is unnecessary to have a detector which selectively detects only one of the radioelements. The specimen and a standard for each of the N radioelements are each counted under N different conditions. Let:

A = the count rate of a standard of radioelement (a)

B = the count rate of a standard of radioelement (b)

C = the count rate of a standard of radioelement (c)

K = the count rate of the specimen

x = the fraction of the standard of (a) in the specimen

y = the fraction of the standard of (b) in the specimen

z = the fraction of the standard of (c) in the specimen.

In the specimen, the sum of individual count rates = the total count rate.

$$A_1x + B_1y + C_1z + \cdots = K_1 \text{ (at condition 1)}$$
$$A_2x + B_2y + C_2z + \cdots = K_2 \text{ (at condition 2)}$$
$$A_3x + B_3y + C_3z + \cdots = K_3 \text{ (at condition 3)}$$
$$\cdot + \cdot + \cdot + \cdots = .$$

The solution to these simultaneous equations gives the fraction of the activity of each standard which is present in the specimen. For only one radioelement (x) in the specimens:

$$x = \frac{K_1}{A_1}$$

For two radioelements (x and y) in the specimen:

$$x = \frac{K_1B_2 - K_2B_1}{A_1B_2 - A_2B_1}$$

$$y = \frac{K_1A_2 - K_2A_1}{B_1A_2 - B_2A_1}$$

For three radioelements (x, y and z) in the specimen:

$$x = \frac{K_1(B_2C_3 - B_3C_2) + K_2(B_3C_1 - B_1C_3) + K_3(B_1C_2 - B_2C_1)}{A_1(B_2C_3 - B_3C_2) + A_2(B_3C_1 - B_1C_3) + A_3(B_1C_2 - B_2C_1)}$$

$$y = \frac{K_1(A_2C_3 - A_3C_2) + K_2(A_3C_1 - A_1C_3) + K_3(A_1C_2 - A_2C_1)}{B_2(A_2C_3 - A_3C_2) + B_2(A_3C_1 - A_1C_3) + B_3(A_1C_2 - A_2C_1)}$$

$$z = \frac{K_1(A_2B_3 - A_3B_2) + K_2(A_3B_1 - A_1B_3) + K_3(A_1B_2 - A_2B_1)}{C_1(A_2B_3 - A_3B_2) + C_2(A_3B_1 - A_1B_3) + C_3(A_1B_2 - A_2B_1)}$$

These equations are well adapted for evaluation by digital computers. There is no theoretical limit to the number of radioelements which may be evaluated in this manner. However the amount of radioactivity and the efficiency and resolution of the detector impose practical limits on the number of radioelements which can be simultaneously resolved. It is basically the simultaneous solution of N linear equations in N unknowns. The trick is in knowing beforehand which radioelements are present in the specimen.

REFERENCES

1. D. H. Taysum, G. D. Westenskow, C. W. Mays, M. A. Van Dilla, A. G. Liddell and L. W. Blake, *K-9 Bark II, An Improved Live Dog Gamma Counter.* Semi-annual progress Report, Radiobiology Laboratory, University of Utah, AECU-3583, 105–109 (1957).
2. M. A. Van Dilla, R. L. Schuch and E. C. Anderson, *Nucleonics* **12** (9), 22 (1954).
3. E. C. Anderson, R. L. Schuch, J. D. Perrings and W. H. Langham, *Nucleonics* **14** (1), 26 (1956).
4. M. A. Van Dilla, B. J. Stover, R. L. Floyd, D. R. Atherton and D. H. Taysum, *Rad. Res.* **8,** 417 (1958).
5. B. J. Stover and D. R. Atherton, *Rad. Res.* In press.
6. W. R. Eckelmann, J. L. Kulp and A. J. Schulert, *Science* **127,** 266 (1958).
7. P. F. Gustafson and C. E. Miller, *Bremsstrahlung Measurement of* Sr^{90} *in Dogs.* Argonne National Laboratory Quarterly Report, ANL-5518, 49–51 (1956).
8. M. A. Van Dilla, R. L. Floyd, J. S. Arnold and B. J. Stover, *Strontium Retention and Bone Metabolism Experiments* Annual Progress Report, Radiobiology Laboratory, University of Utah, AECU-3418, 24–33 (1955).
9. R. L. Floyd, D. H. Taysum and M. A. Van Dilla, *Strontium Retention and Bone Metabolism Experiments*—II. Annual Progress Report, Radiobiology Laboratory, University of Utah, 89–108 (1956).

Health Physics Pergamon Press 1958. Vol. 1. pp. 288–305. Printed in Northern Ireland

MAXIMUM PERMISSIBLE AMOUNTS OF NATURAL URANIUM IN THE BODY, AIR AND DRINKING WATER BASED ON HUMAN EXPERIMENTAL DATA

S. R. BERNARD*
Health Physics Division, Oak Ridge National Laboratory,† Oak Ridge, Tennessee

(*Received* 8 *August* 1958)

Abstract—The distribution of uranium in the human body following intravenous injections of hexavalent and tetravelaent uranium has been studied in the case of eight terminal patients. At the dosage levels used the data indicate that the kidneys and the skeleton are the principal sites of deposition with approximately equal amounts in each. On this basis the kidneys become the critical organ and the toxic effect of the uranium rather than radiation damage becomes the limiting effect in determining the maximum permissible concentrations (m.p.c.) for occupational exposure. The influence of particle size on retention of inhaled material is considered in interpreting some of the available human data.

INTRODUCTION

A SPECIFIC shortcoming in the maximum permissible amounts in the body, q, in the air man breathes, $(\text{m.p.c.})_a$ or m.a.c., and in his drinking water, $(\text{m.p.c.})_w$, recommended by the National Committee on Radiation Protection[1] (NCRP), or the International Commission on Radiological Protection[2] (ICRP), or by AEC supported laboratories,[3] long recognized by these official bodies and also by workers in the field as well as others, is the inherent uncertainty in the values resulting from extrapolation of laboratory animal data to man. Extrapolation to man, using data obtained from controlled laboratory experiments with small research animals, a practice performed of necessity for many self-evident reasons, suffers primarily because of lack of knowledge as to how the extrapolation should be adequately performed. Because of this the values lack the desired certainty.

Some reduction in the uncertainty was achieved during investigation of the chemical toxicity of uranium on hospital patients by BASSETT *et al.*[4] Complete urinary uranium excretion studies, tests of renal function and blood studies were made and from these results the injected uranium dose which produces minimal damage to the kidney tubule of man was determined. However, autopsy data were not obtained since none of the subjects were terminal patients. Consequently, the uncertainty, in so far as the distribution of uranium was concerned, was not reduced or could not even be determined. However, late in 1953 an experiment to determine the distribution and excretion of uranium in man was initiated and the data obtained are believed to be an initial step, and a major one in overcoming the aforementioned difficulty.

Human biopsy and autopsy data on terminal patients have been made available in a careful experiment on the distribution and excretion of uranium in man,[5–7] in a co-operative study by the Health Physics Division, Oak Ridge National Laboratory and the Department of Neurosurgery, Massachusetts General Hospital. Uranium solutions, enriched with isotopes of U^{235} or U^{233}, were administered intravenously to patients who were in terminal stages of irreversible central nervous system disease but who were otherwise in generally good physical condition. From measurements of uranium

* Present address: Committee on Mathematical Biology, The University of Chicago, 5741 Drexel Avenue, Chicago 37, Ill.

† Operated by Union Carbide Nuclear Company for the United States Atomic Energy Commission.

found in biopsy and autopsy tissues the parameter values characterizing the distribution and excretion of uranium in man were determined. In addition, the burden of uranium in man to produce a definite nephrotoxic syndrome was determined from clinical tests on blood, from tests of renal function and from pathological examination of kidney tissues.

The purpose of this report is to present these terminal patient data together with the human parameter values derived from careful consideration of the data and the values for maximum permissible amounts, viz. q, $(\text{m.p.c.})_a$, and $(\text{m.p.c.})_w$ based on these findings. While the number of cases observed is small and the conclusions somewhat indeterminate because of the rather large variability found, still the general paucity of human data warrants the careful consideration of any results suggested by these data. It is concluded from considerations of the terminal patient data, patient data of BASSETT *et al.*,[4] and some data on industrial workers, that the primary hazard from chronic exposure to compounds of natural uranium regardless of their solubility, is due to chemical toxicity on the kidneys. These data would indicate that the safe burden in the kidney, dictated by considerations of chemical toxicity, is one-tenth the burden deemed permissible from radiological considerations. Also, at these dose levels, the human kidneys seem to deposit and retain as much uranium as does the skeleton. Therefore, the biological half-life in the kidneys seems to be the same as that in the skeleton. This finding was not evidenced in small research animals. If the biological half-life is increased, the $(\text{m.p.c.})_a$ and $(\text{m.p.c.})_w$ for soluble compounds is lower than that recommended by the ICRP and NCRP. For the case of insoluble compounds it is suggested that on the basis of dissolution and subsequent absorption of uranium from the lungs, the maximum permissible lung burden based on radiation exposure is too high; consequently, the $(\text{m.p.c.})_a$ for insoluble compounds of natural uranium is also too high. In order to determine the $(\text{m.p.c.})_a$ it was assumed, necessarily, that the retention and distribution of inhaled particulates appearing in Table C. III of the ICRP,[8] apply.

Specific data on the fate of inhaled uranium aerosols of different particle sizes are lacking for small animals, let alone human data, and consequently, the maximum permissible concentration values are not certain. It is believed that this inadequacy of information is due in part to the lack of an adequate hypothesis from which an experiment could be designed to obtain information for determining maximum permissible concentrations as a function of particle size distribution. A hypothesis, derived from considerations of a few small animal data obtained by LABELLE,[9] which takes particle size and its effect upon deposition and lung clearance into consideration, is advanced in this report. This concept has been applied to the animal inhalation data obtained by the University of Rochester and to some lung autopsy measurements of two industrial workers reported by EISENBUD and QUIGLEY.[10] The results of these applications are presented and discussed. Some specific suggestions and recommendations for experiments to obtain data pertinent to this problem are also presented.

SOLUBLE COMPOUNDS

Summary of terminal patient data

A summary of those results found in the experiment on the distribution and excretion of uranium in man pertinent to the problem of setting m.p.c.'s follows. The details of the experiment may be found elsewhere.[5–7]

Eight terminal brain tumor patients, six of which were comatose, whose average age was 45 years (they ranged from 26 to 63 years) were intravenously administered prepared solutions of uranium. Six patients were injected with hexavalent uranium while the other two were administered tetravalent uranium as uranium tetrachloride. The injections ranged from 4 to 50 mg of uranium.

Hexavalent uranium was found to clear rapidly from the blood into the urine. Sixty nine per cent of the injected dose was excreted in the first day (Table 1). Uranium excretion approaches the power function law (Figs. 1 and 2)[6] after about 10 hr post injection.

Autopsy measurements reveal that hexavalent uranium chiefly stores in bones and kidneys while tetravalent uranium prefers storage in

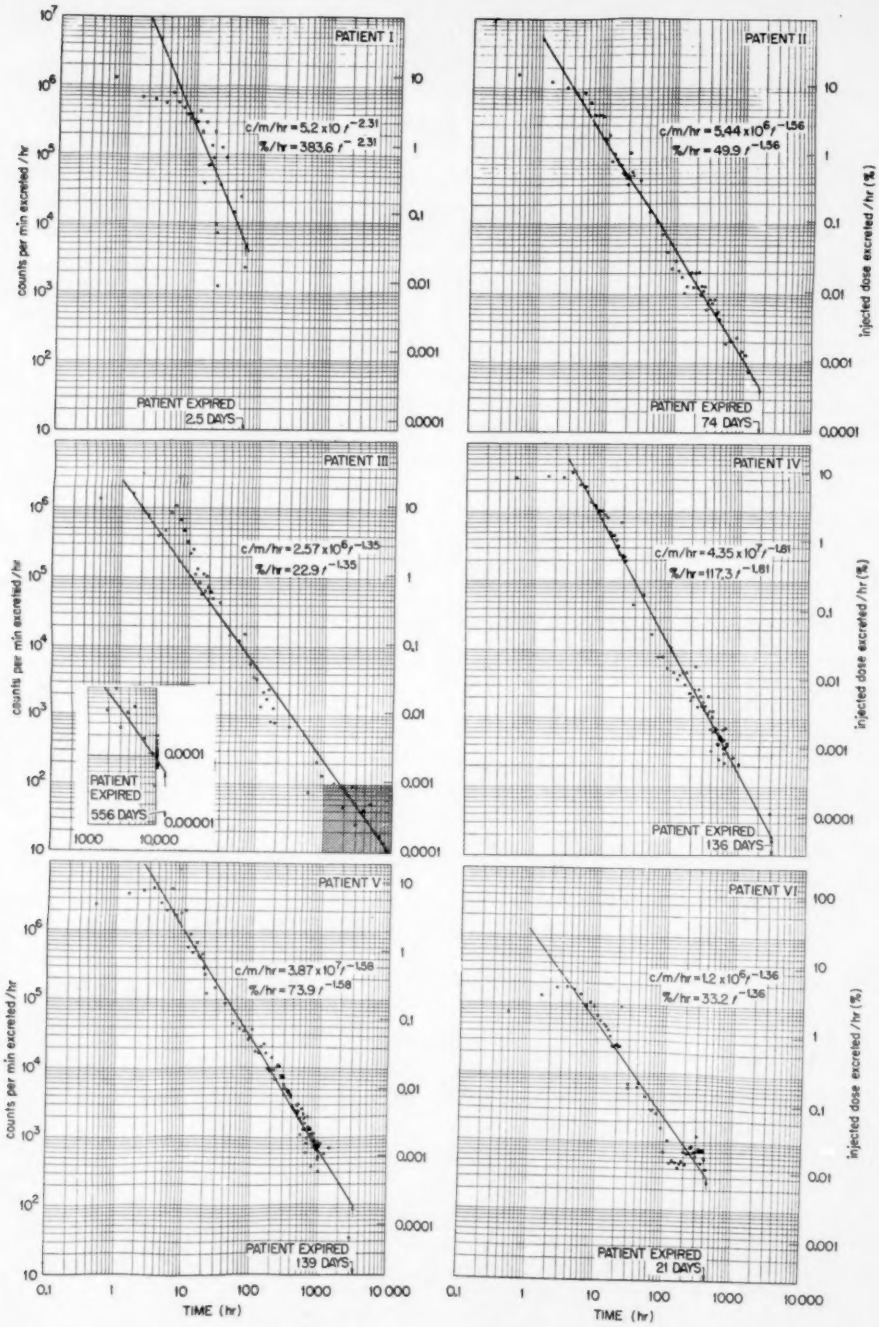

PATIENT I
c/m/hr = 5.2 x 10 t^-2.31
%/hr = 383.6 t^-2.31
PATIENT EXPIRED 2.5 DAYS
PATIENT II
c/m/hr = 5.44 x 10^6 t^-1.56
%/hr = 49.9 t^-1.56
PATIENT EXPIRED 74 DAYS
PATIENT III
c/m/hr = 2.57 x 10^6 t^-1.35
%/hr = 22.9 t^-1.35
PATIENT EXPIRED 556 DAYS
PATIENT IV
c/m/hr = 4.35 x 10^7 t^-1.81
%/hr = 117.3 t^-1.81
PATIENT EXPIRED 136 DAYS
PATIENT V
c/m/hr = 3.87 x 10^7 t^-1.58
%/hr = 73.9 t^-1.58
PATIENT EXPIRED 139 DAYS
PATIENT VI
c/m/hr = 1.2 x 10^6 t^-1.36
%/hr = 33.2 t^-1.36
PATIENT EXPIRED 21 DAYS
counts per min excreted / hr
injected dose excreted / hr (%)
TIME (hr)

FIG. 1. Power function excretion equations.

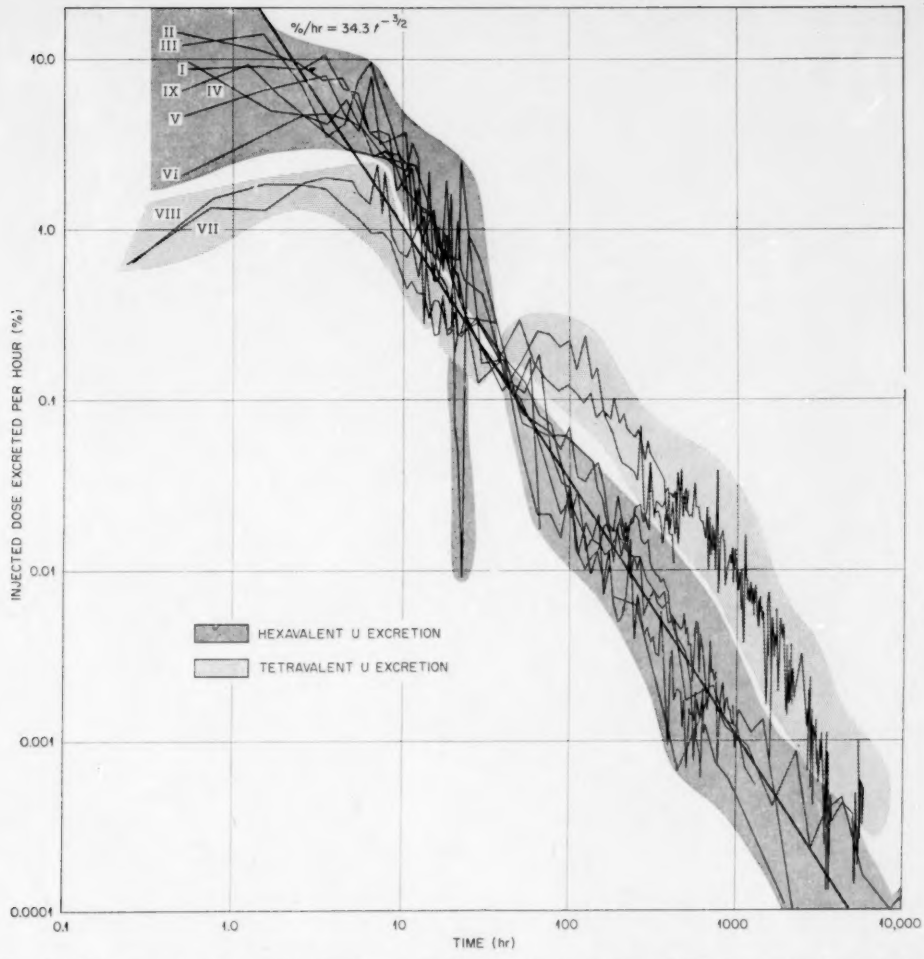


FIG. 2. Urinary rate of excretion for 8 terminal brain tumor patients following intravenous injection.

Table 1. *Percentage of injected dose excreted in first* 24 hr *collection of urine*

Patient*	I	II	III	IV	V	VI	VII	VIII
	59.4	78.0	83.8	77.2	66.5	49.1	20.0	16.9
Average			69.0				18.5	

* Patients I–VI injected with hexavalent uranium.

liver and bones (Table 2). Also, the uranium burdens in bone and kidneys for practical purposes are not essentially different from each other; they contain nearly identical amounts of uranium. This means the biological half-life for bones is the same as that for kidneys.

The biological half-life for bones is different for each patient (Fig. 3). It increases in proportion to $t^{4/5}$ where t is the time of expiration measured from the time of injection (Fig. 4). Fitting a single exponential term to the bone autopsy data by the usual least-squares procedure yields the equation $0.03 \exp(-0.693t/280)$ with t in units of days (Fig. 5, curve 1) which curve poorly fits the data and is inadequate because the biological half-life depends on the expiration time. A better fit of these bone

Table 2. Percentage of injected dose per standard man organ or tissue for six terminal brain tumor patients

Organ of tissue	Sample amount (g)	Patient no.					
		I	VI	II	V	III	VIII
		Expiration time (days)					
		2.5	18	74	139	566	21
Bone	7000	10.0	4.9	1.4	0.6	1.3	14.4
Kidney	300	16.6	7.2	0.7	1.2	0.4	1.1
Muscle	30,000	1.2	2.1	0.9	0.3	0.06	0.4
Skin and subcutaneous tissue	6100	1.8	1.0	0.1	0.06		
Fat	10,000	0.6	0.6			0.04	
Red marrow	1500			0.02	0.03	0.1	
Blood	5400	1.0	0.2	0.005	0.002	0.004	0.08
Stomach	250	0.08	0.02	0.003	0.001	0.001	
Small intestine	1100	0.2	0.2	0.03	0.01	0.006	0.1
Liver	1700	1.8	1.1	0.2	0.2	0.05	9.2
Brain	1500						
Lungs	1000	0.5	0.4	0.03	0.02	0.008	0.3
Heart	300	0.06	0.02	0.003	0.006	0.002	0.004
Spleen	300	0.6	0.2	0.1	0.02	0.006	5.6
Urinary bladder	150	0.03		0.002	0.001	0.0003	0.06
Pancreas	70	0.7	0.008	0.008	0.0006	0.0004	
Testes	40		0.01	0.008	0.002	0.002	0.008
Thyroid gland	20		0.003	0.0002	0.0001	0.0002	0.0009
Prostate gland	20		0.003	0.0004	0.0004	0.0001	0.003
Adrenal gland	20	0.02	0.01	0.003	0.001	0.0004	0.02
Miscellaneous tissues (blood vessels, cartilage, nerves, etc)	390	0.3	0.2	0.04	0.002	0.002	0.04
Total in tissues		35	18	4	3	2	
Total excreted in urine		69	63	92	85	98	

Patients I, VI, II, V and III injected with $UO_2(NO_3)_2 \cdot 6H_2O$; Patient VIII injected with UCl_4.

autopsy data is given by the equation 0.25 exp $(-1.19t^{1/5})$ (Fig. 5, curve 2). In order to determine a half-life with which to calculate m.p.c.'s the equation 0.25 exp $(-1.19t^{1/5})$ was integrated from $t = 0$ to $t = 70$ years and the area given by this integral is found to be identical to the integral of 0.11 exp $(-0.693t/300)$ (Fig. 5, curve 3) over the same time interval. Thus, from these considerations it is said that the biological half-life for uranium in bone is 300 days and it is the same in kidneys since kidneys contain the same burden as the bone.

In the study of chemical toxicity of uranium, LUESSENHOP *et al.* found that a body burden of 0.1 mg/kg of body weight produces a definite nephrotoxic effect.

Power function equations were fitted to the body burden measurements (Fig. 6) and to the urinary excretion measurements (Fig. 2). Excretion measurements made in the first 10 hr were omitted from the calculations. The

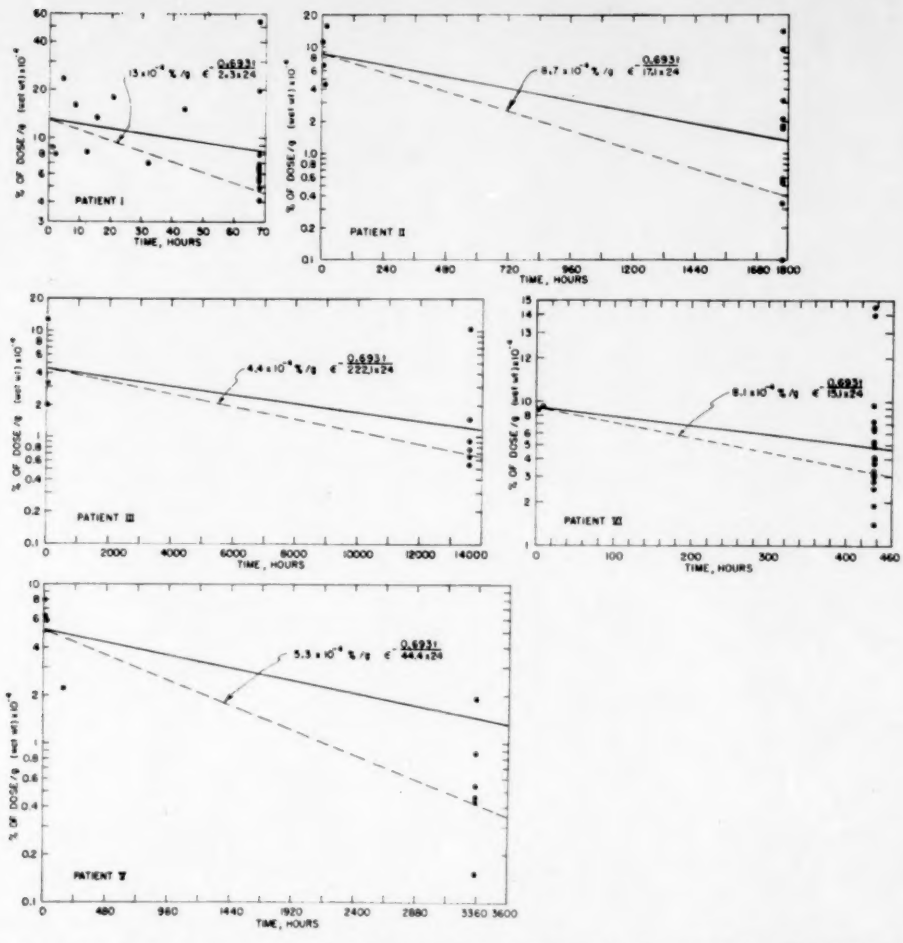


FIG. 3. Best fitting exponential curves for U VI bone biopsy and autopsy samples. The broken curve is that for compact bone samples. The solid curve is for all bone samples.

best fitting equation for body burden is $0.6t^{-1/2}$. The equation for organ burden in bone (or in kidney) is $0.20t^{-1/2}$. Hence, f_2, the fraction in the critical organ relative to that in the total body is $\frac{1}{3}$. The best fitting power function excretion equation is 34.3 per cent/hr $t^{-3/2}$, where t is in hours.

Comparison of terminal patient data with other data

The results of this study with the terminal patients are not in accord with those results obtained from small animals.[2,3] In small animals the principal organ for hexavalent uranium storage is bone; 85 per cent of the uranium in the body resides in bone. The data presented here suggest that in humans there are two principal storage organs, bones and kidneys; 33.3 per cent of the uranium residing in each organ. The biological half-life in kidneys of small animals is 30 days and f_2', the fraction going from blood to kidneys is 0.33. In humans the biological half-life is 300 days and f_2' is 0.11.

In general, there was very little, if any, difference exhibited in the urinary excretion of hexavalent uranium by patients of BASSETT *et al.* (Rochester patients). This is surprising in view of the fact that their clinical states were different. None of the Rochester patients had

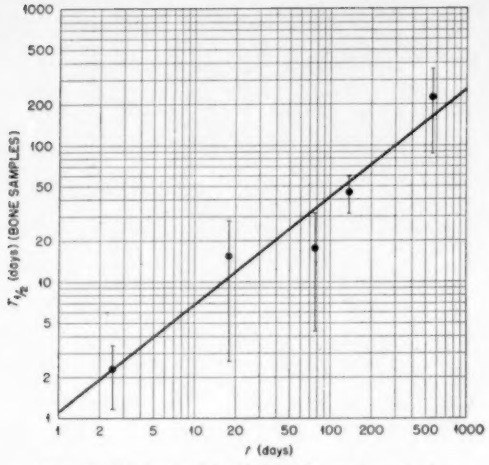


FIG. 4. Biological half-life of uranium in bone plotted vs. expiration time.

maladies comparable in severity to the terminal patients (Boston patients). There was a slight difference in experimental technique with respect to: (a) sample collection (Rochester patients voided at natural times whereas Boston patients were catheterized); and (b) level of injection dose (Rochester patients were administered smaller (0.4–4 mg) doses of uranyl nitrate hexahydrate). The urinary excretion findings were similar in these respects: (a) the best fitting power function excretion equation, 57.2 per cent/hr $t^{-1.8}$ (Fig. 7) differed slightly from that of Boston patients, viz. $34.3t^{-1.5}$; and (b) the amount of uranium excreted in the first day was essentially the same, 76 per cent as compared with 69 per cent.

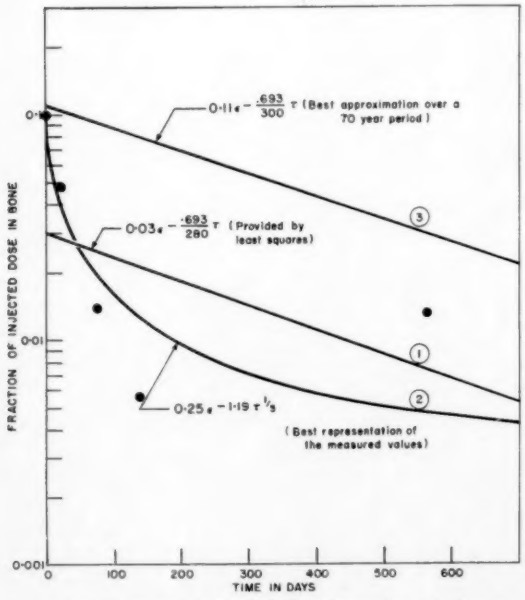


FIG. 5. Biological half-life of uranium in bone.

Maximum permissible body burden, q

Based on chemical toxicity. NEUMAN[12] recommended a safe kidney burden of 2–3 μg/g of kidney for chronic exposure. Since NEUMAN's recommendation (1948) additional human data have been made available upon which to base a safe burden in kidney.

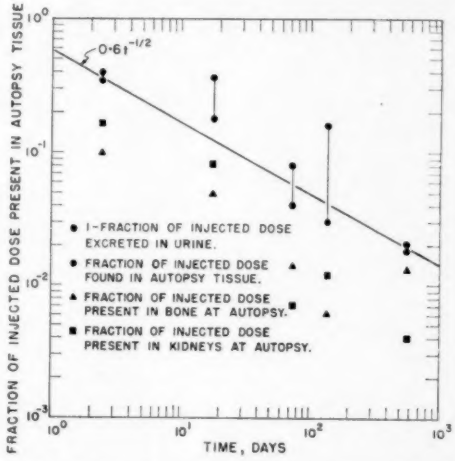

$0{\cdot}61t^{-1/2}$
● 1-FRACTION OF INJECTED DOSE EXCRETED IN URINE.
● FRACTION OF INJECTED DOSE FOUND IN AUTOPSY TISSUE.
▲ FRACTION OF INJECTED DOSE PRESENT IN BONE AT AUTOPSY.
■ FRACTION OF INJECTED DOSE PRESENT IN KIDNEYS AT AUTOPSY.
FRACTION OF INJECTED DOSE PRESENT IN AUTOPSY TISSUE
10^0
10^{-1}
10^{-2}
10^{-3}
10^0 10^1 10^2 10^3
TIME, DAYS

FIG. 6. Uranium burden in organs and body.

In the experiments with the terminal patients, the value of LUESSENHOP *et al.*, 0.1 mg/kg body weight, assuming a 70 kg man and 300 g of kidney and one-third of the dose in kidney, corresponds to 6 μg/g of kidney.

EISENBUD and QUIGLEY[10] have urinary excretion data from five individuals who received a short massive exposure to uranium hexafluoride. Abnormal urinary findings were said to have persisted for a few days but in all cases they returned to normal, indicating only minimal effects on the kidneys despite the relatively high exposures encountered. The

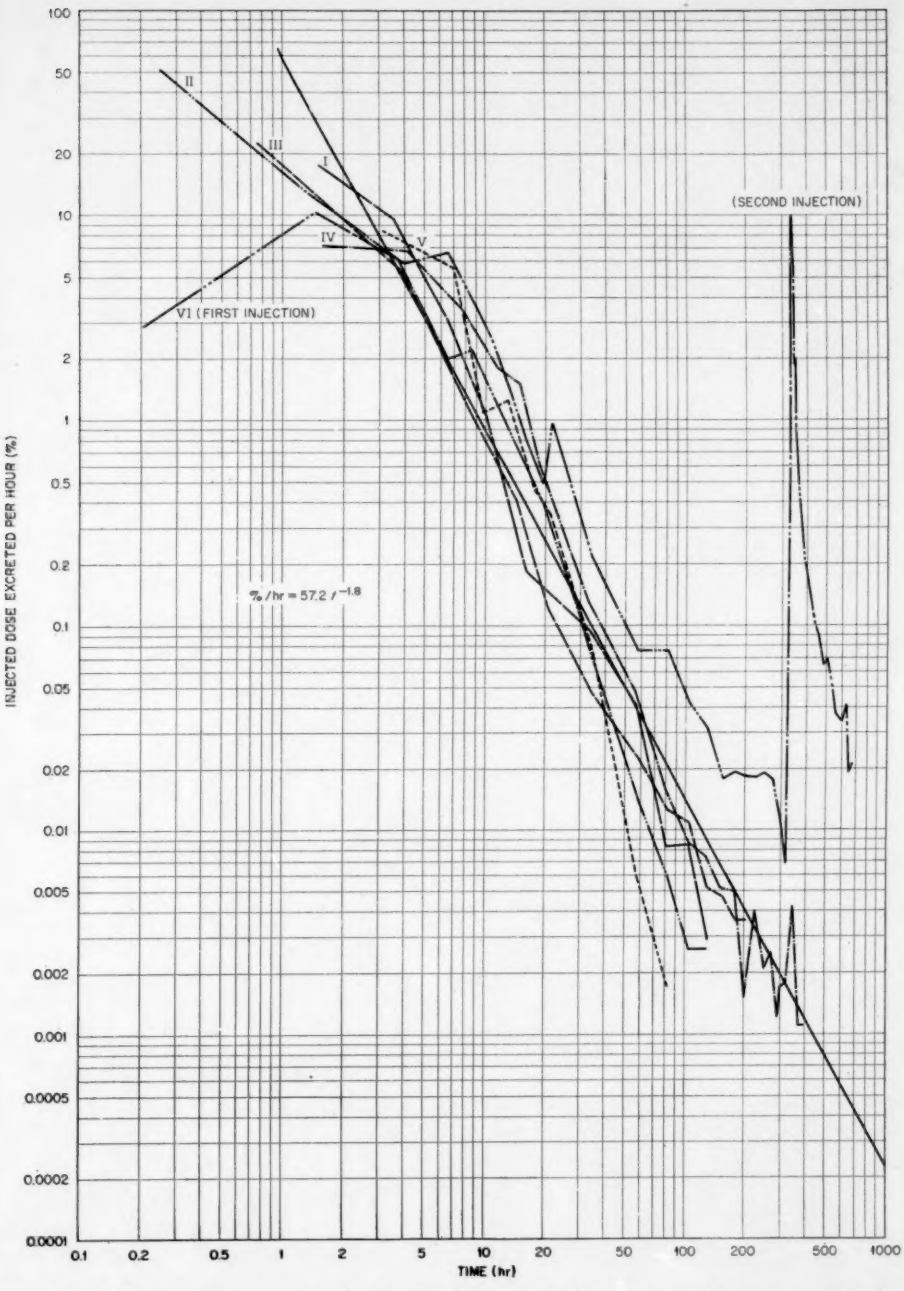


FIG. 7. Urinary rate of excretion for 6 patients at the University of Rochester, following intravenous injection.

average concentration of uranium in the first days urine of these individuals was 1.28 mg/l. of urine. Assuming 1 l. of urine excreted per day, 69 per cent of the uranium introduced into the blood-stream excreted in the first day, one-third of this amount is in the circulating blood and 300 g of kidney, then this excretion level is found to correspond to 2 μg/g of kidney.

The sixth Rochester patient, studied by BASSETT *et al.*, was administered two intravenous injections of uranyl nitrate hexahydrate, the first of which contained 3.1 mg and the second, injected 17 days after the first, contained 3.1 mg. BASSETT *et al.* concluded tolerance had been reached after the second injection because they noted a slight rise in urinary catalase and protein. These injections correspond to a kidney burden of 4 μg of uranium per g of kidney assuming one-third of the second injection deposited in kidney and that the amount present in the kidney from the first injection is given by the equation $0.20It^{-1/2}$, which equation fits the kidney data of the terminal patients. (In the equation I is the injection dose. The time t is in units of days.)

The average of these three values, 2, 4 and 6 μg/g of kidney is 4 μg/g of kidney. This is slightly different from NEUMAN's recommendation of 2–3 μg/g of kidney as a safe burden for chronic exposure and assuming 300 g of kidney and one-third stored in the body resides in kidney, then the safe burden is 2700 μg of U_{nat}. In terms of radioactivity this is equivalent to 0.0009 μc* of U_{nat}. Thus, from the standpoint of chemical toxicity, the permissible body burden is 0.0009 μc U_{nat}.

Based on radiological considerations and the human data. When the human parameter values are used to calculate qf_2, the burden to give 0.3 rem/week to the kidney, it is found that

$$qf_2 = 8.4 \times 10^{-4} m/\Sigma E(r.b.e.)N = 0.0026 \ \mu\text{c } U_{nat},$$

where $m = 300$ g and $\Sigma E(r.b.e.)N = 94$ MeV.[2] Setting $f_2 = \frac{1}{3}$ and solving for q yields $q = 0.008$ μc U_{nat}. Thus, on the basis of radiation exposure, when the body contains 0.008 μc U_{nat}, the kidney, the critical organ receives 0.3 rem/week. However, 0.008 μc of U_{nat} is nine times that deemed permissible from chemical toxicity considerations. Because of this large factor, it is necessary to accept the value, 0.0009 μc U_{nat}, based on chemical toxicity considerations.

Maximum permissible concentration in air, $(m.p.c.)_a$, based on chemical toxicity and human data

When the $(\text{m.p.c.})_a$ is calculated with either the exponential law or the power function law retention it is found that they give essentially the same values, viz. 1×10^{-12} μc U_{nat}/cm^3 of air.

$$(\text{m.p.c.})_a = \begin{cases} \dfrac{3.5 \times 10^{-8} qf_2}{Tf_a(1 - \exp(-0.693t/T))} \\ \qquad = 1.3 \times 10^{-12} \ \mu\text{c } U_{nat}/cm^3 \\ \dfrac{5 \times 10^{-8} q}{f_a' \displaystyle\int_0^t 0.6(t-\tau)^{-\frac{1}{2}} d\tau} \\ \qquad = 1.1 \times 10^{-12} \ \mu\text{c} U_{nat}/cm^3. \end{cases}$$

where $q = 0.0009$ μc, $f_2 = \frac{1}{3}$, $f_2' = 0.11$, $T = 300$ days, $f_a' = (0.25 + f_1)f_2'$, $f_1 = 10^{-4}$, $t = 1.825 \times 10^4$ days (50 years) and $f_a' = f_a/f_2'$. In the equations it is to be noted that human parameter values are used in all cases except for the values of f_1 and f_a' (fraction of inhaled particulates absorbed into the bloodstream) which values were obtained from ICRP.[13] There are no human experimental measurements of these latter values and this adds to the uncertainty of $(\text{m.p.c.})_a$.

Maximum permissible concentration in water $(m.p.c.)_w$, based on chemical toxicity and human data

When the $(\text{m.p.c.})_w$ is calculated with either the exponential law or the power function law for retention it is found that they yield nearly identical values, 5.8×10^{-6} μc/cm^3, and 5.1×10^{-6} μc/cm^3, respectively. The equations

* 1 μc U_{nat} = 1 μc U^{239} + 1 μc U^{234} + $\frac{9}{370}$ cμ U^{235}

for computing these values are:

$$(\text{m.p.c.})_w = \begin{cases} \dfrac{3.15 \times 10^{-4}\, q f_2}{T f_w (1 - \exp(-0.693 t/T))} \\ \qquad = 5.8 \times 10^{-6}\ \mu\text{c/cm}^3 \\ \dfrac{4.54 \times 10^{-4} q}{f_1 \displaystyle\int_0^t 0.6(t-\tau)^{-\frac{1}{2}}\, d\tau} \\ \qquad = 5.1 \times 10^{-6}\ \mu\text{c/cm}^3 \end{cases}$$

where $q = 0.0009\ \mu$c, $f_2 = \frac{1}{3}$, $f_w = f_1 f_2' = 5.5 \times 10^{-5}$, $f_1 = 5 \times 10^{-4}$, $f_2' = 0.11$ and $t = 1.825 \times 10^4$ days (50 years) and $T = 300$ days. For practical purposes the value $6 \times 10^{-6}\ \mu$c of U_{nat} per cm^3 of water is selected as the $(\text{m.p.c.})_w$. It is to be noted that the human parameter values are used in the calculations except in the case of f_1, which value was obtained from ICRP,[13] also. Because of the many uncertainties involved the $(\text{m.p.c.})_w$ cannot be considered as determined.

Comparison with other recommendations

The values for q and the m.p.c.'s prescribed from considerations of the human data and chemical toxicity have been found to be substantially lower than those recommended by ICRP or NCRP or the University of Rochester. However, the values recommended by these three bodies have been found to be substantially different from each other, viz. the current ICRP recommendation for occupational exposure (40 hr/week) to soluble compounds of U_{nat} is 270 μg/m^3[14] while NCRP recommendation corresponds to 76 μg/m^3[15] and the University of Rochester recommends 50 μg/m^3[16] (the period of exposure is not cited but it is generally assumed that this concentration is for 40 hr/week). The $(\text{m.p.c.})_a$ based on chemical toxicity and the human data, for occupational exposure, is 9 μg/m^3 which value is a factor of 30 lower than ICRP' recommendation and a factor of 5 lower than the University of Rochester's recommendation.

The basis for the 270 μg/m^3 (this value was suggested by the Harriman, New York, Conference, April 1953) is not indicated by ICRP.[17] Consequently, the reasons for the difference between ICRP value and the NCRP value are not known. Regardless of the reasons, it can be shown that the kidney burden resulting from exposure to 270 μg/m^3, 40 hr/week, assuming the exponential law for retention and employing small animal parameter values, would be 21 μg/g of kidney tissue, seven times higher than the safe burden of 3 μg/g of kidney. Therefore, the ICRP recommendation is believed to be in error.

The basis for the NCRP recommendation is the small animal data obtained by investigators at the University of Rochester. The small factor of difference (1.5) between the NCRP' value, 76 μg/m^3, and the Rochester value, 50 μg/m^3, is due to the assumption, by NCRP, that Rochester's recommendation was for an 8 hr day, 7 days/week, exposure period.[18] A careful examination of the record of the exposure periods for small animals exposed to dusts consisting of soluble compounds of U_{nat} reveals that the animals were exposed for an average of 28 hr/week.* Hence, the 50 μg/m^3 level corresponds to an exposure period of 28 hr/week. If it can be said that animals, continuously (168 hr/week) exposed, would store six times more uranium than that observed and that the toxicological response from continuous exposure would be six times more pronounced, then the $(\text{m.p.c.})_a$ would have to be 9 μg/m^3.

On the basis of these considerations of the small animal findings, the 9 μg/m^3, which is equivalent to $3 \times 10^{-12}\ \mu$c/cm^3, is only a factor of 3 greater than the value recommended on the basis of human data and chemical toxicity.

---

* For example, dogs and rats received 1600 hr of exposure in a period of 12 months to four different levels, 0.04, 0.25, 0.40 and 2.0 mg/m^3[19] of $UO_2(NO_3)_2$, while rats and guinea pigs were exposed for 900 hr.[19] These exposure periods represent 31 and 26 hr/week, respectively. Also, dogs and rats were exposed for 1680 hr while rabbits and guinea pigs were exposed for 1200 hr to UF_6 at two different levels, 0.05 and 0.20 mg/m^3.[20] These exposure periods correspond to 33 and 23 hr/week. In addition, dogs and rats were exposed for 1700 hr,[21] rabbits and guinea pigs for 1000 hr,[21] to UCl_4, at two different levels, 0.05 and 0.20 mg/m^3,[21] for 1 year. These exposure periods correspond to 33 and 19 hr/week.

The average of all the above exposure periods is 28 hr/week.

INSOLUBLE COMPOUNDS

Preliminary considerations of the values of q and $(m.p.c.)_a$ recommended by ICRP and/or NCRP

There are no human data from controlled experiments with which to evaluate the m.p.c.'s for insoluble compounds of U_{nat}. Nevertheless, some elementary considerations of the dissolution of lung stored uranium and its absorption into the bloodstream, lead to the conclusion that the current recommendations are a factor of 10 too high.

Currently, the ICRP recommends $q = 0.01$ μc[14] for insoluble compounds, based on lungs as the critical organ. Consider, the amount entering the bloodstream per day, $\lambda q f_2$, due to dissolution of uranium in lung and its absorption in blood:

$$\lambda q f_2 = 6 \times 10^{-5} \ \mu c/day$$

where λ, the effective biological decay constant for lungs is 0.693/120 days^{-1}, while $q = 0.01$ μc and $f_2 = 1$. This amount, 6×10^{-5} μc/day, is ten times higher than the amount entering the bloodstream from daily inhalation of the $(m.p.c.)_a$ of soluble compounds of U_{nat} which is $2 \times 10^7 \ (m.p.c.)_a \ f_a' = 5 \times 10^{-6}$ μc/day, where $(m.p.c.)_a = 1 \times 10^{-12}$ $\mu c/cm^3$ and $f_a' = 0.25 = f_a/f_2'$. Thus, the value of q based on radiological considerations, is a factor of 10 too high. Therefore q, on the basis of these chemical toxicity considerations, should be 0.001 μc.

Because of the above considerations, it is believed that the current $(m.p.c.)_a$, 3×10^{-11} $\mu c/cm^3$, prescribed by ICRP, for insoluble compounds of U_{nat}, is also high by a factor of ten. Therefore, it is recommended that the $(m.p.c.)_a$ for insoluble compounds, should be 3×10^{-12} $\mu c/cm^3$.

Comparison with other recommendations

In experiments with the small animals, University of Rochester investigators found that an exposure period of 33 hr/week[22] to an insoluble compound, UO_2, at a dust level of 50 $\mu g/m^3$, resulted in a lung deposition and retention of 12 $\mu g/g$ of lung, after 1 year of exposure. From these data, and on the basis that 25 $\mu g/g$ of lung[23] corresponds to 0.3 rem/week (α-particle r.b.e. = 10), they recommended levels of 50–100 $\mu g/m^3$[23] for application in the field. Again they did not indicate the period of exposure but it may be presumed that it is for 40 hr/week.

Taking into consideration that continuous exposure of these animals could be expected to lead to a lung burden five times greater than that observed, then the level of exposure, 50 $\mu g/m^3$, could be expected to result in a burden of 60 $\mu g/g$ of lung for the case of continuous exposure, 24 hr/day, 7 days/week. Expressing these μg values in terms of μc yields 1×10^{-12} $\mu c/cm^3$ which is equivalent to a lung burden of 1×10^{-6} $\mu c/g$ of lung. Assuming 1000 g of lung, the $(m.p.c.)_a$ for insoluble compounds of U_{nat} therefore, would be 1×10^{-12} $\mu c/cm^3$.

Eisenbud and Quigley[10] have autopsy data on two industrial workers who died from non-occupational causes and were exposed to high concentrations of UF_4 and UO_2 for periods of 1 or 2 years. One worker was exposed for 2 years to an average concentration of 17,000 $\mu g/m^3$ and 10 months later he died and the lung burden measured 0.35 $\mu g/g$. Another worker died 15 months after exposure for a year to 5,000 $\mu g/m^3$ and his lungs measured 0.23 $\mu g/g$. There were no measurements listed for kidneys or bones or lymph glands from these workers.

Eisenbud predicted, from the small animal data, that the worker exposed for 2 years would have 600 $\mu g/g$ while the worker exposed for 1 year would have 56 $\mu g/g$. Because of the large difference between the predicted burden and the measured burden, he questioned the application of animal results directly to man without knowledge as to comparative relation of deposition and clearance rates in the two species. He concludes that the safety factor in the $(m.p.c.)_a$ for insoluble compounds is unnecessarily conservative but that more data are needed before an upward revision in the $(m.p.c.)_a$ could be permitted.

It is agreed, more human lung autopsy measurements are needed, but in addition to the measured lung burdens there is need for measurements of the particle sizes inhaled by the workers. It is believed that the large discrepancies can be explained in terms of the

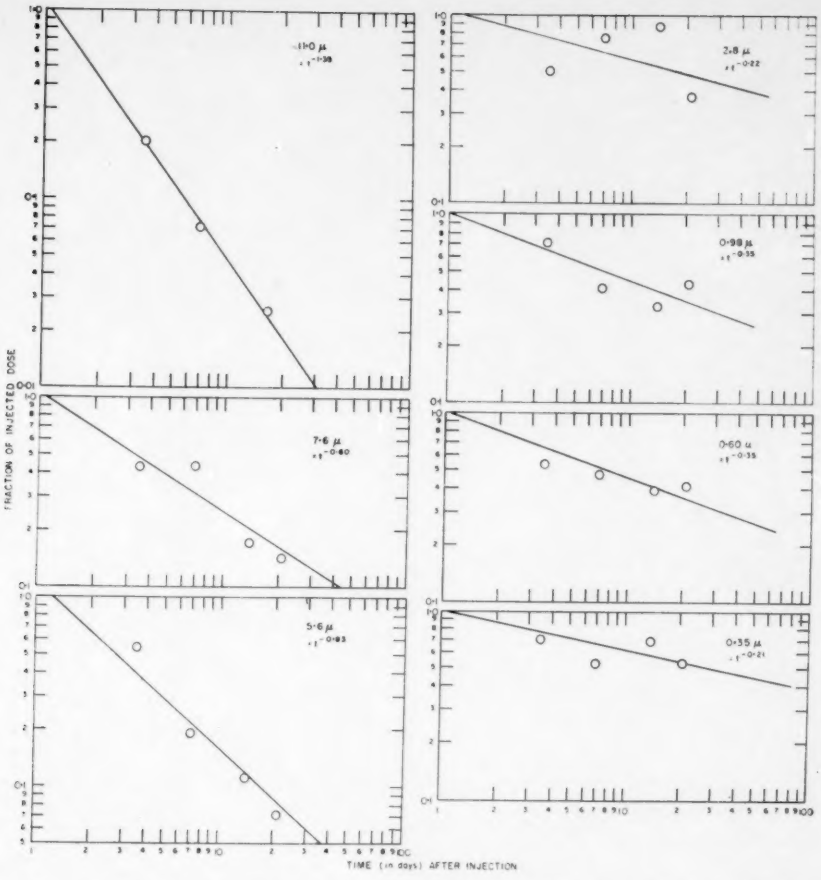


FIG. 8. Time retention of U in lungs of rats after intratracheal injection of various sizes of UO_2 dust particles.

particle size as the following considerations show.

Particles size effect on lung retention and clearance

LABELLE,[9] in an effort to understand the effect of particle size on the toxicological response shown by animals, provided some data which go a long way in understanding the effect of particle size on the retention and clearance of uranium from the respiratory tree. In his experiment, LABELLE intra-tracheally injected slurries of UO_2, composed of different sized particles into rats. He killed these rats at 3, 7, 14 and 21 days after injection and measured the burden in lungs, kidneys and femur. LABELLE graded the oxide particles by an elutriation technique, that of CUMMINGS,[25] and obtained seven different sizes of particles, ranging from 11 to 0.35 μ in diameter. He then injected each rat with a dose of 30 mg/kg of body weight of the seven graded sizes and placed the rats in cages where they stayed until the time of sacrifice.

Log x log graphs of the fraction of the injected uranium plotted versus the time of sacrifice reveals smaller particles are eliminated more slowly from the lungs than larger size particles (Fig. 8). Furthermore, the retention data can be approximately represented with the power function, At^{-n}, while A, the intercept at $t = 1$

day, is unity and n, the exponent is a function of particle diameter, x. An approximate relationship between the exponent of the power function and the particle diameter is $n(x) = 0.26x^{-0.56}$(Fig. 9).

From the above considerations, a quantitive relationship between retention in lungs at any time t after deposition of a unit amount of uranium is:

$$R(t, x) = t^{-n(x)} \tag{1}$$

where, $R(t, x)$ is the amount present at time t after injection of particles of diameter x, and where:

$$n(x) = 0.26x^{-0.56} \tag{2}$$

In equations (1) and (2) t is in units of days and x is in units of μ.

A concept for determining $(m.p.c.)_a$ as a function of particle size

If the concentration in the air is denoted by $(\text{m.p.c.})_a$, then the amount deposited in the lungs during a time interval $d\tau$ at time τ is $2 \times 10^7(\text{m.p.c.})_a D(x)\, d\tau$ where $D(x)$ is the fraction of the inhaled aerosol deposited in alveolar passageways. The burden in the lung at a later time t, is then:

$$q(x, t) = 2 \times 10^7(\text{m.p.c.})_a \int_0^t D(x)(t - \tau)^{-n(x)}\, d\tau \tag{3}$$

Performing the indicated integration in equation (3) yields:

$$q(x, t) = 2 \times 10^7(\text{m.p.c.})_a D(x) t^{1-n(x)}/(1 - n(x)) \tag{4}$$

Equation (4) was solved for the $(\text{m.p.c.})_a$ to give unit lung burden after 10^{-1}, $\frac{1}{2}$, 1, 2, 3 and 50 years of continuous exposure, assuming $D(x) = 1.22\{\exp(-0.9x) - \exp(-2.3x)\}$, which equation approximately represents the alveolar retention data of HATCH and HEMEON.[26] A graph of the $(\text{m.p.c.})_a$ to give unit lung burden as a function of particle size shows that large concentrations of large particles are required to maintain unit lung burden (Fig. 10).

Equation (4) is not adequate for application to the case of a distribution of particle sizes. However, it can be made adequate by multiplying by $\frac{4}{3}\pi\rho x^3 P(x)$, where ρ is the density of the particle, assumed to be spherical, $P(x)$ is the probability distribution of sizes whose mean size is $\bar{x}$ and whose standard deviation is s, and integrating the resultant expression over all

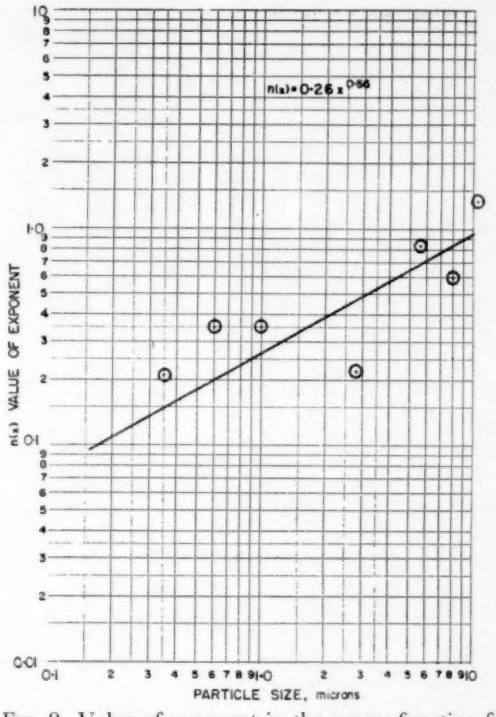


FIG. 9. Value of exponent in the power function for different particle sizes of UO_2.

sizes, and dividing by a normalization factor, the integral over all sizes of $\frac{4}{3}\pi\rho x^3 P(x)\, dx$. In equation form this is:

$$q(\bar{x}, s, t) = \frac{\int_0^\infty x^3 P(x) q(x, t)\, dx}{\int_0^\infty x^3 P(x)\, dx} \tag{5}$$

where $q(\bar{x}, s, t)$ is the lung burden after continuous exposure to an aerosol composed of a distribution of particle sizes. The denominator of equation (5) is said to be the third moment of the distribution, its value depending upon the choice of distribution. In many practical situations the particles in an aerosol can be characterized by the log-normal distribution. The

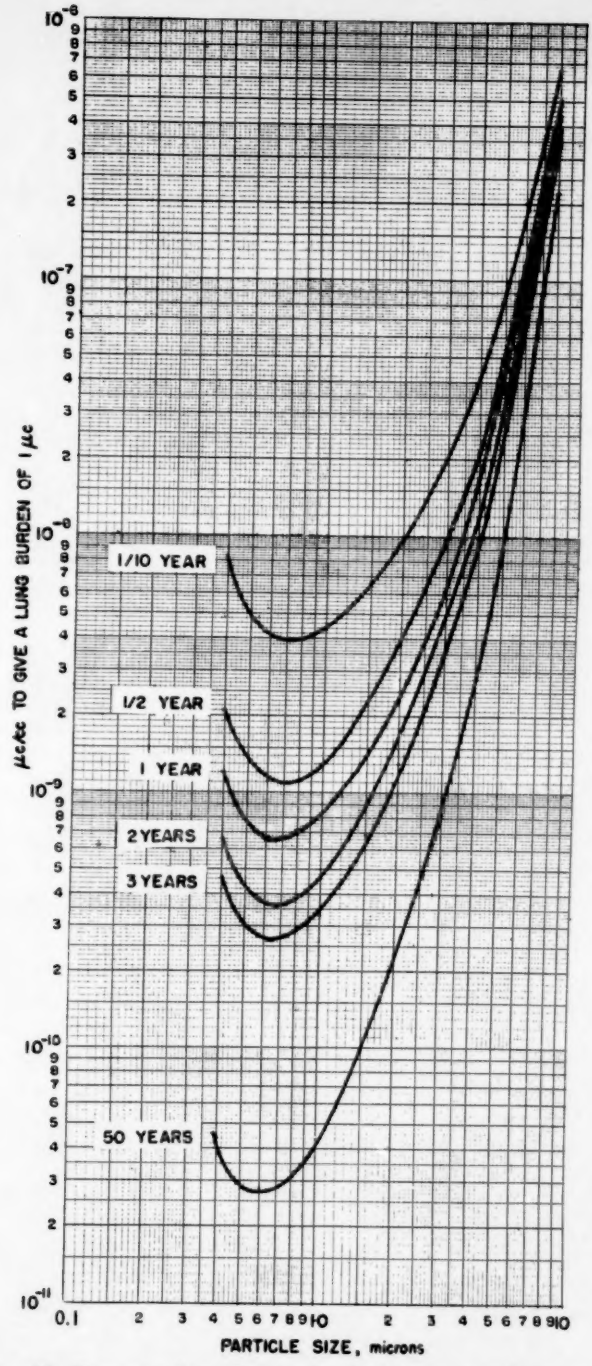


FIG. 10. Concentration of aerosols of different particle size to give a lung burden of 1 μc for periods of continuous exposure specified.

parameters for a log-normal distribution are x_g, the geometric mean and s_g, the geometric standard deviation. The equation for the log-normal distribution is:

$$P(x) = \frac{1}{\log s_g \sqrt{(2\pi)}} \times \exp\left\{-\tfrac{1}{2}\frac{(\log x - \log x_g)^2}{\log s_g}\right\} \quad (6)$$

and the third moment of this distribution is: $x_g^3 \exp(4.5 \log^2 s_g)$.[27] Thus, the lung burden resulting from inhalation of an aerosol whose particle sizes are log-normally distributed is given by:

$$q(x_g, s_g, t) = \bar{x}_g^3 \exp(-4.5 \log^2 s_g) \int_0^\infty x^3 P(x) q(x, t)\, dx \quad (7)$$

Application of concept to animal inhalation data and human autopsy data

When equation (7) was applied to the lung burden measurements obtained in the inhalation experiments with small animals performed by the University of Rochester a considerable overestimate of the burden was evidenced. In part this overestimate is due to an incorrect assumption, namely, the alveolar retention data of HATCH and HEMEON apply to uranium dusts. In experiments with uranium oxide of 2.6 μ and 0.45 μ, WILSON *et al.*[28] found 0.9 per cent and 8.3 per cent of the inhaled dose, respectively, retained or more precisely, deposited in the alveolar passageways. In recent experiments performed in this laboratory, it was found that 10 per cent of inhaled enriched uranium fumes, whose mean particle size is 0.36 μ, is deposited in lower respiratory passages of dogs. Thus, more of the finer material was deposited, and a smaller amount of the uranium dusts are retained. Because of this, the equation for $D(x)$, which is for the HATCH and HEMEON data, was divided by a factor of 10.

In the inhalation experiments performed on dogs and rats by the University of Rochester, 1 and 10 mg/m^3 levels of uranium dioxide dust whose geometric mean particle size was 1.3 μ,[29] and geometric standard deviation was 2.3μ,[29] furnished the exposure. Both species were exposed for 1 year. The lung burden was predicted with equation (7) for this distribution of particles and these levels of dust, assuming that dogs breathe 2.5 m^3/day and 100 g of dog lung, and taking into consideration the fact that they received one-fifth of a continuous exposure. For rats, 0.1 m^3 daily air intake and 1 g of lung was assumed. The predicted lung burden was slightly less than that observed for dogs and about a factor of 10 greater than that measured in rats (Fig. 11a and 11b).

In the experiments with UF_4 dust, at 3.0 and 0.5 mg/m^3, $x_g = 1.84$[30] and $s_g = 2.43$,[30] the predicted lung burdens computed with equation (7) were greater, but by not more than a factor of 10, than the measured lung burdens for rats and dogs (Fig. 12a and 12b).

For application to EISENBUD and QUIGLEY's autopsy data, the average air concentration was assumed to be for occupational exposure and the geometric mean particle size = 2.4 and geometric standard deviation = 2.4 was also assumed. In addition it was necessary to take into consideration the fact that these workers' exposures ceased before they died, thereby allowing for elimination of uranium from lungs during conditions of non-exposure. (For discussion of the manipulation of the power function for application to the case of continuous exposure followed by a non-exposure period, the reader may consult the paper by HEALY.[31]) Under these conditions the predicted lung burdens were 800 and 100 μg/g of lung, little different from the extrapolated values determined by EISENBUD and QUIGLEY, viz. 600 and 56 μg/g, but extremely different from the measured burdens, 0.35 and 0.23 μg/g. On the other hand, had these workers been exposed to particles whose mean diameter, $x_g = 6$ μ and whose standard deviation = 2.0 μ, then the predicted lung burdens would be 9.0 and 0.3 μg/g of lung.

DISCUSSION

Conservativeness of $(m.p.c.)_a$, $(m.p.c.)_w$ and q

The values for q, $(\text{m.p.c.})_a$ and $(\text{m.p.c.})_w$ based on the human data and chemical toxicity are lower than those recommended by others and therefore they may be said to be conservative. These m.p.c. values are lower for two

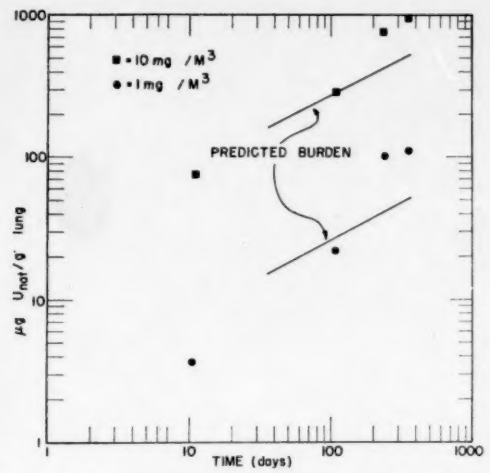


FIG. 11(a). Comparison between predicted and measured lung burdens of dogs exposed to UO_2 dust at levels of 1 and 10 mg U/m³

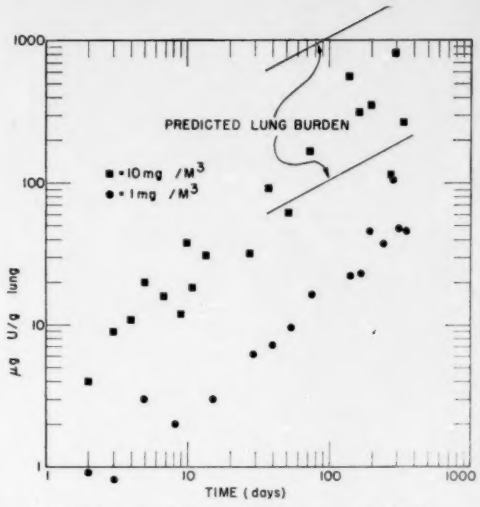


FIG. 11(b). Comparison between predicted and measured lung burdens of rats exposed to UO_2 dusts at levels of 1 and 10 mg U/m³.

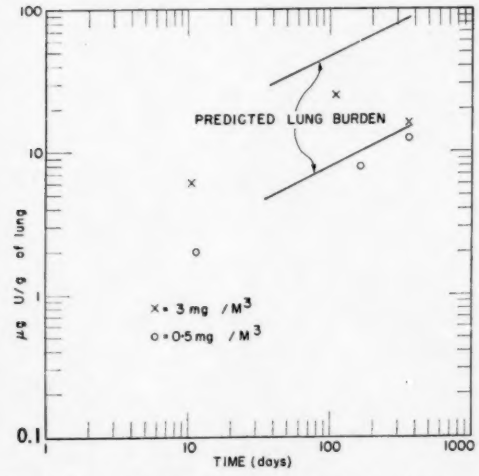


FIG. 12(a). Comparison between predicted and measured lung burdens of dogs exposed to UF_4 dust at levels of 0.5 and 3.0 mg U/m³.

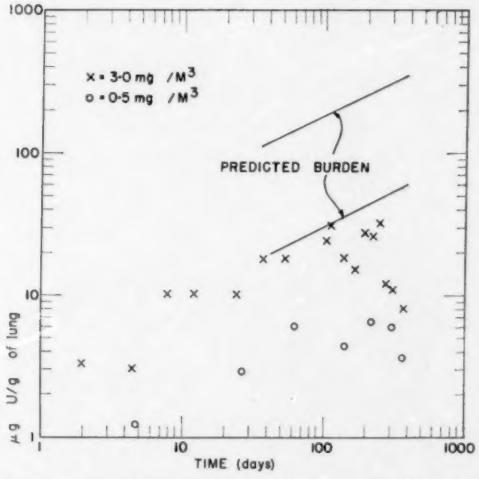


FIG. 12(b). Comparison between predicted and measured lung burdens of rats exposed to UF_4 dusts at levels of 0.5 and 3.0 mg U/m³.

reasons; (a) human kidneys are considered to retain more uranium and to store it for longer periods of time and (b) these m.p.c. values are for continuous and chronic exposure. These points require elaboration.

There is some element of doubt as regards the application of the data obtained from terminal brain tumor patients to normal subjects. If it were known that normal subjects' kidneys did not concentrate uranium to the extent indicated somewhat higher m.p.c. values would be in order. Some indirect evidence, the agreement between excretion patterns demonstrated by the Rochester patients and the terminal patients, leads to the belief that the tissue distribution is the same. Admittedly however,

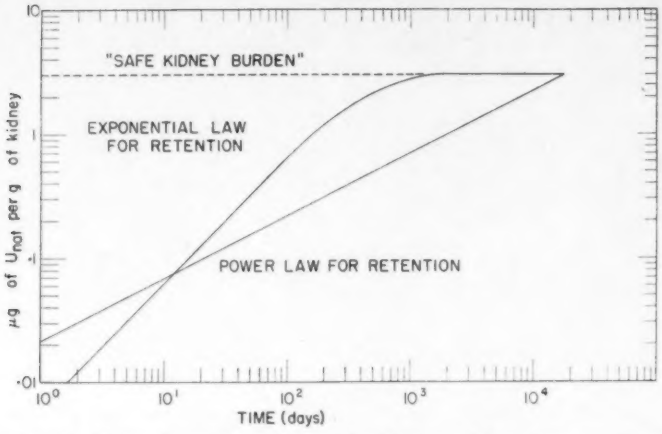


FIG. 13. Natural uranium kidney burden resulting from continuous inhalation of 1×10^{-12} μ/cc of U_{nat}.

this evidence is inadequate and data from normal subjects is required before ultimate confidence can be placed in the terminal patients findings. Moreover there is evidence from animal experiments of a dose effect, and that at lower levels the relative size of the kidney burden may be smaller.

The m.p.c. values, applicable to the case of chronic exposure, have been designed such that the kidney burden at the end of 50 years of continuous exposure will not exceed the safe level (Fig. 13) after termination of exposure. The power law predicts that after 50 years of exposure the kidney burden will be maintained at an essentially constant level—the decrease will be less than a factor of 2—even though the exposure has been terminated for 20 years. In this light the m.p.c.'s do not appear to be conservative.

Inadequacy of data for determination of $(m.p.c.)_a$ as a function of particle size

The data obtained by LABELLE on lung retention of various size particles are inadequate for determining $(\text{m.p.c.})_a$ for various distributions of particle sizes. This was evidenced in the application to the inhalation data. The agreement between the predicted and measured lung burdens is fortuitous, perhaps, since LABELLE injected slurries of oxide particles and saline into the rats lungs. The effect upon the clearance mechanisms by the saline vehicle is not known. Hence, these data cannot be said to be representative of the clearance of particulates from the respiratory tree which were deposited upon inhalation. However, a modified procedure would yield useful data.

Suggested experiment to obtain data for determination of $(m.p.c.)_a$ as a function of particle size

An experiment which would yield the necessary data would consist of the following:

(1) Prepare slurries of oxide particles of different sizes by LABELLE's elutriation technique.

(2) Disperse these slurries in a chamber similar to the chamber of WILSON *et al.*

(3) Allow experimental subjects to inhale the homogeneous size particulate matter from the chamber through a mask and a special valve to direct the air flow.

(4) After the inhalation exposure collect excreta samples in addition to samples of lung, bone and kidney and measure these for uranium.

(5) Plot the data and determine the parameters of distribution and excretion of inhaled uranium as a function of the particle size comprising the aerosol inhaled.

Acknowledgments—The author gratefully acknowledges the assistance of those members associated with "project Boston" of ORNL at the time of this writing. In addition, he would like to acknowledge the

assistance given by FLORENCE HOUSCH LUKEMAN, CAROL S. BANKS, WENDELL B. NIX and J. CROWELL.

REFERENCES

1. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, National Bureau of Standards, Handbook 52. (1953).
2. *Recommendations of the International Commission on Radiological Protection*, British Journal of Radiology, Supplement No. 6, London (1955).
3. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds*, Pt. 1 and 2. McGraw-Hill, New York (1949); Pt. 3 and 4. McGraw-Hill, New York (1953).
4. S. H. BASSETT, A. FRENKEL, N. CEDARS, H. VAN ALSTINE, C. WATERHOUSE and K. CUSSON, *UR-37* (1948).
5. E. G. STRUXNESS, A. J. LUESSENHOP, S. R. BERNARD and J. C. GALLIMORE, *The Distribution and Excretion of Hexavalent Uranium in Man. International Conference on the Peaceful Uses of Atomic Energy* (1955).
6. S. R. BERNARD, J. R. MUIR and G. W. ROYSTER, JR., *The Distribution and Excretion of Uranium in Man. Proceedings of the Health Physics Society* (1956).
7. S. R. BERNARD and E. G. STRUXNESS, *A Study of the Distribution and Excretion of Uranium in Man.* ORNL-2304 (1957).
8. *Recommendations of the International Commission on Radiological Protection*, British Journal of Radiology, Supplement No. 6, p. 26, London (1955).
9. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* pp. 1730–1778. McGraw-Hill, New York (1953).
10. M. EISENBUD and J. A. QUIGLEY, *Industrial Hygiene of Uranium Processing. International Conference on the Peaceful Uses of Atomic Energy* Vol. XIII, pp. 222–228. United Nations, New York (1956).
11. A. J. LUESSENHOP and J. C. GALLIMORE. Unpublished report.
12. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 2247. McGraw-Hill, New York (1953).
13. *Recommendations of the International Commission on Radiological Protection*, British Journal of Radiology, Supplement No. 6, Table C, IV, p. 30, London (1955).
14. *Recommendations of the International Commission on Radiological Protection*, British Journal of Radiology, Supplement No. 6, Table C. VIII, p. 53, London (1955).
15. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, National Bureau of Standards, Handbook 52, Table 3, p. 14.
16. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 2241. McGraw-Hill, New York (1949).
17. *Recommendations of the International Commission on Radiological Protection*, British Journal of Radiology, Supplement No. 6, Note (d), p. 54, London (1955).
18. K. Z. MORGAN, Personal communication.
19. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 2362. McGraw-Hill, New York (1953).
20. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 2364. McGraw-Hill, New York (1953).
21. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 2366. McGraw-Hill, New York (1953).
22. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 1553. McGraw-Hill, New York (1953).
23. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 2410. McCraw-Hill, New York (1949).
24. M. EISENBUD, Retention, *Arch. Industr. Hyg. Occup. Med.* **6,** 214 (1952).
25. D. E. CUMINGS, *J. Ind. Hyg.* **11,** 245 (1929).
26. G. DRINKER and T. HATCH, *Industrial Dust* (2nd Ed.) pp. 93–94. McGraw-Hill, New York (1954).
27. G. HERDAN, *Small Particle Statistics.* Elseviev, New York (1953).
28. H. B. WILSON, G. E. SYLVESTER, S. LASKIN, C. W. LABELLE and H. E. STOKINGER, *Arch. Industr. Hyg. Occup. Med.* **6,** 93 (1952).
29. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* p. 1561. McGraw-Hill, New York (1953).
30. C. VOEGTLIN and H. C. HODGE, *Pharmacology and Toxicology of Uranium Compounds* pp. 1929–1930. McGraw-Hill, New York (1953).
31. J. HEALY. *Proposed Revision in Computation of MPC's for Long-lived Radioisotopes.* HW-35663 (1955).

Health Physics Pergamon Press 1958. Vol. 1. pp. 306–314. Printed in Northern Ireland

HISTORY OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL UNITS AND MEASUREMENTS (ICRU)

LAURISTON S. TAYLOR*

(*Received* 31 *March* 1958)

Abstract—The International Commission on Radiological Units and Measurements was established in 1925 by the first International Congress of Radiology and since that time has established the units that are used in radiation dosimetry. The roentgen was introduced in 1928 and the rad in 1953. Since 1953 the Committee's activities have broadened substantially and in its most recent studies provided much of the technical information and background necessary for the practical interpretation of radiation dose in terms of the rad. This report covers the work of the Commission as it has developed since its formation over 30 years ago.

1. FORMATION

THE ICRU was formed in 1925 under the auspices of the First International Congress of Radiology, then meeting in London. The single factor having the most bearing on the formation of this Commission was the absence of any units of radiation dosage for use in therapeutic application of radium and X-rays. Up to that time, several units had been proposed and were in use in various countries but there was no international acceptance of any one of them.

2. MODE OF OPERATION AND HISTORY

In its initial formation, the ICRU consisted of two representatives from each of the countries participating in the Congresses of Radiology. Of these two members, one was expected to be a physicist and one a radiologist. In practice, about three-quarters of the members were primarily radiologists. Meetings of the ICRU were held during the International Congress. The continuity of the Commission and the arrangement of meetings, agenda, etc., were in the hands of a secretary selected from among the Commission members. The meetings of the Commission were presided over by one or two honorary chairmen who were usually people of substantial scientific or radiological reputation from the country in which the meeting was held. Occasionally there were also honorary secretaries having no defined responsibilities. In general, these honorary positions were filled by persons having little or no contact with the radiological field and hence were honorary in the strictest sense of the word.

Since there were of the order of fifty countries participating in each International Congress, the rules of the Commission permitted as many as a hundred members, but many countries made no attempt to designate members and the largest number meeting at any one time was forty-one. With such a large membership, it was extremely difficult to carry out the Commission's work; it usually developed that some eight or ten persons carried the load. On the other hand, many countries felt they should have a voice in the matter, with the result that there was an enormous amount of useless discussion at each of the meetings.

Formal minutes of the meetings were not kept. It was considered that the report released by the Commission constituted its findings as well as its minutes.

1925

At the first ICRU in London, a series of papers relating to X-ray dose measurement were presented.[1] Following the discussion of these papers a resolution to study the question of X-ray measurement was passed, and a committee was appointed to nominate an

* Chairman, ICRU 1953–1959.

international committee for this purpose.[2] The nominating committee included the following:

Sir W. Bragg
Prof. F. H. Hopwood
Dr. E. A. Owen
Mr. C. E. S. Phillips
Prof. A. W. Porter
Prof. S. Russ

1928

The second meeting of the ICRU was held in Stockholm in 1928 at which time there were about forty members present. At this meeting, the Committee adopted the definition of a unit of "X-ray intensity" known as the roentgen.[3] This was an extremely important step forward and for the first time it was made possible to measure radiation in all countries in terms of the same unit. This unit has continued in use to the present time with various modifications in definition, but only minor changes in magnitude.

A resolution adopted in 1928 included the statements:

"(1) That this international unit be the quantity of X-radiation which, when the secondary electrons are fully utilized and the wall effect of the chamber is avoided, produces in one cubic centimeter of atmospheric air at 0°C and 76 cm mercury pressure, such a degree of conductivity that one electrostatic unit of charge is measured at saturation current.

"(2) That the international unit of X-radiation be called 'the Röntgen', and that it be designated by the small letter 'r'."

The membership of the ICRU in 1928 is not known.

1931

The third meeting of the ICRU was held in Paris in 1931 at which time thirty-nine members were present.[4] The following recommendations were made:

(1) The International Unit of X-radiation shall be the quantity which, when the secondary electrons are fully utilized and the wall effect of the chamber is avoided, produces in 1 cm^3 of atmospheric air at 0°C, and 76 cm mercury pressure, such a degree of conductivity that one electrostatic unit of charge is measured at saturation current.

(2) The International Unit of X-radiation shall be called the "roentgen" and shall be designated by the letter "r".

(3) The "intensity" of the radiation shall be expressed in r/sec.

Between the 1928 and 1931 meetings, international comparisons of X-ray standards were carried out, one in 1928 by H. Behnken, using a portable thimble chamber, and one in 1931 by L. S. Taylor using a portable primary standard. As a result of these intercomparisons, the ICRU, in 1931, recommended that the National Standardizing Laboratories establish agreement on the general characteristics of primary X-ray standards and of the radiation quality ranges over which these standards were to be used.[5] This then marked the first time that there was substantial agreement regarding the basic characteristics of primary radiation standards. The Committee further recommended that:

(1) The experimental methods of establishing a standard for the determination of the International X-ray Unit shall be entrusted to a subcommittee consisting of the following members of the Units Committee: M. de Broglie (France), W. Friedrich (Germany), E. A. Owen (Great Britain), R. Sievert (Sweden), I. Solomon (France), E. Pugno-Vanoni (Italy), L. S. Taylor (U.S.A.); (Honorary Secretary of the Committee, E. A. Owen). This Committee shall invite the collaboration of the various existing national bureaus for standard measurements and also those about to be instituted.

(2) This Committee shall consider: (a) methods of controlling the constancy of dosage meters; (b) the correlation of X-ray and γ-ray dosage; (c) the establishment of a γ-ray unit of intensity.

(3) The progress of the work done by the subcommittee shall be reported once a year to the members of the International X-ray Unit Committee.

(4) Each country shall be requested immediately to elect its two representatives on the International X-ray Unit Committee; until new representatives are elected the present members shall serve.

(5) The International Committee shall

Date and place of meeting	Members present	Officers
1925, London	?	E. A. Owen (U.K.), Secretary
1928, Stockholm	40	E. A. Owen (U.K.), Hon. Secretary H. Holthusen (Germany), Secretary
1931, Paris	39	E. A. Owen (U.K.), Hon. Secretary H. Holthusen (Germany), Secretary
1934, Zurich	33	E. A. Owen (U.K.), Hon. Secretary H. Holthusen (Germany), Secretary
1937, Chicago	41	L. S. Taylor (U.S.), Secretary
1950, London	13	W. V. Mayneord (U.K.), Chairman L. S. Taylor (U.S.), Secretary
1953, Copenhagen	12	W. V. Mayneord (U.K.), Chairman L. S. Taylor (U.S.), Secretary
1956, Geneva	11*	L. S. Taylor (U.S.), Chairman W. J. Oosterkamp (Netherlands), Secretary
1957, Geneva (with ICRP)	6	L. S. Taylor (U.S.), Chairman R. Sievert (Sweden) Co-chairman

* Main Commission only; 11 subcommittee members and 5 subcommittee consultants were also present.

henceforth be called "The International Committee for Radiological Units".

Thus, beginning in 1931, in addition to the members designated by the member countries of the Congress, the Commission also included in its membership representatives designated by each of the recognized National Laboratories. At the time, this included the National Laboratories of England, Germany, Sweden, and the United States.

At the same time, it became evident that the large size of the ICRU made its deliberations unwieldy. It was therefore proposed that the continuity of program and the direction of the technical discussions be placed in the hands of a small group known as the Executive subcommittee. This arrangement was put into effect at the following meeting.

1934

The fourth meeting of the ICRU was held in Zurich and St. Moritz in 1934, and was attended by thirty-three members.[6] At the outset of this meeting it was decided to establish an Executive subcommittee for the purpose of formulating the technical program for subsequent consideration by the whole committee. This plan, put into effect at the Zurich meeting, proved to be very effective, and the first set of preliminary operating rules for the ICRU were formulated.* These were as follows:

(1) The International Committee for Radiological Units shall be constituted by two representatives from each country sending delegates to the congress. When a country has a central X-ray standardization laboratory, it may in addition send a representative of that laboratory. Of the representatives from a single country at least one must be a radiologist, and one a physicist.

(2) There shall be a standing Subcommittee of the International Committee for Radiological Units consisting of six members including the chairman.

(3) The subcommittee shall elect its own Chairman and Secretary from amongst its members.

(4) The Subcommittee shall report on the progress of dosage-measurements and prepare the program to be submitted to the Main

* *Proc. IV Int. Cong. Radiol.* **4,** 148 (1934).

Committee. The report shall be published and circulated to members of the Main Committee at least six months before the meeting of the Congress.

The first executive subcommittee consisted of:

I. SOLOMAN (France) Chairman
L. S. TAYLOR (U.S.A.) Secretary
E. A. OWEN (U.K.)
H. BEHNKEN (Germany)
E. PUGNO-VANONI (Italy)
R. SIEVERT (Sweden)

For the first time the ICRU report carried some recommendations relative to the clinical measurement of dose, and the calibration of dose-meters. Also the ICRU recognized the agreements of the National Laboratories on X-ray standards.[7] The section on clinical measurement follows:

(1) Dosage measurement shall be made in roentgens. Dosage rate shall be expressed in r/min.

(2) All dosage measurements shall distinguish between the air dose which does not include scattered radiation, and the effective dose which includes scattered radiation.

(3) The specification of dosage shall include a statement of the quality of the radiation. For exact physical measurements the quality of the radiation may be specified in terms of the complete absorption curves in copper or aluminum. For most practical purposes the quality may be expressed by the first and second half-value layers of the same materials.

(4) The specification of dosage shall also include: (a) dosage rate, (b) the total time of irradiation and the intervals between the times of irradiation, (c) material and thickness of filter, (d) target-skin distance, (e) the dimensions and number of ports of entry.

(5) The practical instrument used to measure X-ray quantity shall be called a dose-meter and shall be calibrated in roentgens.

(6) The calibration of a dose-meter shall be tested periodically by a recognized testing laboratory, over the range of wave length for which it will be used.

(7) The constancy of the calibration of a dose-meter shall be tested by the ionization produced, under fixed conditions by a definite quantity of radium element.

(8) Dose-meters should be independent of the wave length within the range for which they will be used.

The ICR in its Rules and Regulations (plan), confirmed some of the earlier operating procedures of the ICRU and at the same time recognized the formation of the Executive Subcommittee:

The Congress secretariat shall call a meeting of the Committee on Units at the beginning of each International Congress.* The Committee on Units is composed as follows:

(a) An Executive Committee of six members.

(b) Two representatives of all the countries represented at the Congress of whom one shall be a radiologist and the other a physicist.

(c) Delegates of national laboratories for radiological testing.

The Committee on Units chooses its President and its Executive Committee from among its members. A representative of the country in which the Congress is held shall belong to the Executive Committee.

1937

The fifth meeting of the ICRU was held in Chicago in 1937, with forty-one members attending.[8] At this meeting, the Executive Subcommittee was in full operation and the affairs were conducted more expeditiously than ever before.

During the 6 year period from 1931–1937 important technical advances had been made, resulting in the production of X-rays at energies up to 1 MV. This made it apparent that wording of the earlier definition of the roentgen (in reality, more a definition of a device than a physical quantity) would require modification, even though the magnitude of the quantity measured be not changed. It was also decided to make the definition applicable to γ-rays, and accordingly the roentgen was redefined as follows:[8]

(1) The International Unit of quantity or dose of X-rays or γ-rays shall be called the

* *Proc. IV Int. Cong. Radiol.* **4**, 161 (1934).

"roentgen" and shall be designated by the symbol "r".

(2) The roentgen shall be the quantity of X- or γ-radiation such that the associated corpuscular emission per 0.001293 g of air produces, in air, ions carrying 1 e.s.u. of quantity of electricity of either sign.

(3) Measurements of radiation quantity shall be expressed in roentgens. Measurements of dosage rate shall be expressed in roentgens per minute.

While solving some problems, the new definition inadvertently introduced a new source of difficulty, through reference (paragraph 3) to the term "radiation quantity". It is obvious that the roentgen does not measure "radiation quantity" in the usually accepted sense. There had also crept into common usage the acceptance of the roentgen as a measure of "dose"—again a concept at variance with the normally accepted understanding of that term. While not basically affecting the clinical measurement of radiation in the normal energy range, these inconsistencies have plagued the ICRU for many years.

There was developed at this time a more complete pattern for the description of radiation treatment conditions. This was designed to take into consideration all of the many factors necessary for a description of a radiation treatment. These recommendations for recording treatment have been continued since that time but have undergone some modification at almost every meeting of the Committee.

A new and much more detailed set of rules was developed to govern the selection of members and the work of the ICRU.

Thirteen years lapsed before the next meeting, during which time World War II had been in progress, and many of the original members of the Committee had either died or been killed. During this interim period the secretary of the Executive subcommittee, had kept abreast of developments in the field preparatory to reviving the Committee's operations.

In reorganizing the Committee, it seemed advisable to change its basis of membership selection in order to avoid the handicap of working with such large numbers. In advance of the Fifth Congress, discussions held between Dr. A. C. Christie (President of the Fourth ICR), L. S. Taylor and W. V. Mayneord led to agreement that a membership of about twelve would be desirable. With this number, it would no longer be necessary to have an Executive subcommittee. This suggestion was put by the past President of the Fourth Congress to the incoming President of the Fifth Congress, and was tentatively accepted by them subject to later approval by the International Executive Committee of the Congress. Approval was obtained during the 1950 Congress in London, and at the same time the Committee was renamed as a Commission.

1950

The reorganized Commission held its sixth meeting in London in 1950.[9] At this meeting a new set of rules was developed governing the membership and the work of the ICRU. The rules limited the membership of the Commission to a chairman and twelve additional members, selected for their recognized technical ability without regard to nationality. They also insured a reasonable turnover in membership, yet at the same time provided for adequate continuity of membership.

In its 1950 recommendations, the ICRU recognized for the first time the need for absolute measurements of radiation based on calorimetry or other fundamental techniques. It recognized further that the Commission was not in a position at that time to make any specific recommendations but the way was prepared for introducing improvements at a later time.

No changes were made in the 1937 recommendations on the specification of the conditions of X- and γ-ray treatment or the section on instruments. It was recognized at the meeting that some changes were desirable but that the need was not very pressing. Action was deferred until the next Congress.

The definition of the roentgen remained unchanged in spite of some acceptable arguments that change was needed.

The definition of the curie for the measurement of *any* radioactive material was adopted in accordance with the recommendation of the Commission on Standards, Units and Constants

of Radioactivity appointed by ICSU. This made the curie applicable to radioactive materials other than radium.

For the year or two prior to the 1950 meeting, the American and British Units Committees had been very active in the development of the concepts of energy measurements as applied to radiological problems. These were widely circulated well in advance of the meetings with the result that at the 1950 meetings, it was relatively easy to bring about substantial agreement between the different viewpoints prevailing. This advance action paved the way for the acceptance of the new energy unit for dose proposed later by the American and British representatives.

1952

In 1952, a joint meeting was held in Stockholm between the ICRU, the ICRP, and the UNESCO Joint Committee on Radiobiology. This meeting was primarily for the purpose of discussing the genetic aspects of radiation. While it was agreed that no specific recommendations would be made as a result of these meetings, it is interesting to note that the general findings were substantially the same as those made in April 1956 by the ICRP and in June 1956 by the U.S. National Academy of Sciences.

A short meeting of those members of the ICRU present was primarily for discussion of agenda for the regular meeting to be held in 1953.

1953

The seventh meeting of the ICRU was held in Copenhagen in 1953 with twelve members attending.[10] The most important outcome of this meeting was the introduction of a new basic unit for the measurement of radiation dose. This unit, known as the rad, was designed to place the measurement of dose on the basis of first principles. The new concepts and definitions adopted were as follows:

(1) Intensity of radiation is the energy flowing through unit area perpendicular to the beam per unit time. It is expressed in ergs/cm^2 sec^{-1} or W/cm^2.

(2) Quantity of radiation is the time integral of intensity. It is the total energy which has passed through unit area perpendicular to the beam and is expressed in ergs/cm^2 or W-sec/cm^2.

(3) Absorbed dose of any ionizing radiation is the amount of energy imparted to matter by ionizing particles per unit mass of irradiated material at the place of interest. It shall be expressed in rads.

(4) The rad is the unit of absorbed dose and is 100 ergs/g.

(5) Integral absorbed dose is the integration of the energy absorbed throughout a given region of interest. The unit is the gram-rad. 1 g-rad = 100 ergs.

The reasons for adoption of the rad were related (1) to the fact that we were using energies very much higher than was ever dreamed of in the 1930's, and for which the roentgen is not always the most suitable unit of measurement, and (2) the unit of absorbed dose, based on energy imparted to matter, would be applicable to other forms of radiation such as β-rays, α-rays and neutrons. It was clearly recognized during these meetings that much more information needed to be provided before the rad could be regarded as a practical unit. However, since 1953 its attractions became evident and it has begun to appear regularly in the radiological literature.

It will be noted that a physical definition of "quantity of radiation" was included; this mainly to help to distinguish the meaning of the term "quantity" as used in the definition of the roentgen.

Definition of the roentgen was again the subject of extensive discussion, particularly with reference to proposals by the Dutch and British members. While still unhappy about the current definition, and in spite of recognition of the validity of the proposals it was agreed to leave the definition unchanged. It was felt that any gains in physical preciseness of definition might be offset by the feeling among radiologists that their unit of radiation measurement was being changed. In addition, it was felt by some that adoption of the rad would eventually detract from the importance of the roentgen as a radiological unit.

The recommendations on the specification of radiation treatment were extensively revised and

extended to include the technical data required for treatment with radio nuclides.

With some background of experience with the operating rules adopted in 1953 some minor changes were proposed and later approved by the ICR.

The Commission established two subcommittees to provide more concentrated studies in the fields of X-ray standards and standards of radioactivity. W. J. OOSTERKAMP (Netherlands) served as chairman of the Subcommittee on X-ray Standards, and B. RAJEWSKY (Germany) as chairman of the Subcommittee on Standards of Radioactivity.

During the 1953 meetings, the Commission held its first symposium at which invited papers were presented on current work in radiation units and measurements.[11] This provided an opportunity to discuss at open meetings the work being done in various countries and the problems that require further investigation.

1955

In 1955, an informal meeting of the ICRU was held in Geneva during the International Conference on the Peaceful Uses of Atomic Energy. This was attended by the eight members present at the Conference, and was mainly in preparation for the formal meeting to be held the following year.

1956

The eighth meeting of the ICRU was held in Geneva in the spring of 1956. This marked the first time that the Commission had met separately from the meetings of its parent organization—the International Congress of Radiology. Meetings were held jointly with the ICRP and extended over a period of 12 days.

Another departure at these meetings was the fact that they were held with the assistance of the World Health Organization. This resulted from contacts made during the preceding few months, when the WHO had indicated its need for technical advice in the field of radiation protection and units. During the course of the meetings, the Commission officially accepted an earlier invitation of WHO to enter into an official relationship with WHO as a "Non-governmental Participating Organization". As a result of this relationship, the ICRU is now recognized by the World Health Organization as its body of technical advisors in the field of radiological units and measurements.

The meetings were attended by the ten members and the chairman of the Main Commission; eleven subcommittee members and five subcommittee consultants were also in attendance.

The report of the ICRU developed during the 1956 meetings represents the most complete effort thus far.[12] In addition to some degree of clarification of the different units used in measuring radiation dose, the report includes for the first time a large body of technical data called for in the 1950 and 1953 recommendations. It also includes extensive discussions and instructions regarding the problems met in introducing the new energy units into medical and biological practice.

It was agreed that some interim secondary standards of radiation measurement should be developed and made available to any countries requiring calibration of its equipment. The National Bureau of Standards (U.S.) agreed to undertake the responsibility for the development and construction of this equipment, which consists of a spherical cavity ionization chamber, standard diaphragms for free-air chambers and a standardized capacitor for checking current measurements. Through the co-operation of the WHO and UNESCO the secondary standards together with trained personnel will be sent to those countries requiring calibration of their working or primary standards.[13]

The Committee structure of the Commission was reorganized and enlarged; it now consists of the following four Committees:

Committee I:	Standards and Measurement of Radioactivity for Radiological Use. Chairman: W. E. PERRY, U.K.
Committee II:	Standards and Measurement of Radiological Exposure Dose. Chairman: H. O. WYCKOFF, U.S.
Committee III:	Measurement of Absorbed Dose and Clinical Dosimetry. Chairman: L. H. GRAY, U.K.

Committee IV: Standard Methods of Measurement of Characteristic Data of Radiological Equipment and Materials.
Chairman: B. COMBEE, Netherlands

In order to describe better the Commission's scope of activities, it's name was changed to International Commission on Radiological Units and Measurements (ICRU).

During the meetings, two half-day symposia were held for the purpose of presenting and discussing seventeen reports on specialized problems in the field of radiation units and measurements. The symposia were arranged by the ICRU, and through the assistance of the World Health Organization, were held in the UNO buildings. They were attended by some seventy-five persons including members of the ICRP and its subcommittees. The papers were not published as in 1953.

For a period of 3½ days following the Commission meetings, several members (L. S. TAYLOR, W. OOSTERHAMP, L. H. GRAY) of the ICRU met with a WHO study group to consider on a world-wide basis how the recommendations of the ICRU might better be implemented. As one result of this study, the WHO will take active steps to disseminate on a world-wide basis the recommendations developed by the ICRU.

1956

An informal meeting of the ICRU was held in Mexico City in the summer of 1956 during the seventh International Congress of Radiology. Seven members who were in attendance at the Congress met to work on some of the phraseology of the report which had been developed in Geneva.

Listed below are the formal meetings of the ICRU held since its inception, the number of members in attendance, and the Commission Officers. (The Officers listed were in office at the time of meeting. Terms of new officers began after confirmation of their election during the International Congress, and continue for a 3 year period corresponding to the interval between Congresses. Honorary chairmen are not listed, as they were not permanent members of the Commission.)

The Eighth International Congress of Radiology is scheduled to meet in Munich in 1959. The next formal meeting of the ICRU has been scheduled to meet in Geneva in September 1958. It is expected that Committees of the ICRU will meet more frequently than the Main Commission to expedite the completion of their special studies. For the period 1956–1959, the membership of the Main Commission numbers thirteen, and that of the four Committees (including consultants) will be approximately from forty to fifty. Officers of the Main Commission for this period are L. S. TAYLOR (U.S.), Chairman; L. H. GRAY (U.K.), Vice-chairman; H. O. WYCKOFF (U.S.), Secretary.

REFERENCES

1. Discussion on Int. Units and Stds. for X-ray work, *Brit. J. Radiol.* **23,** 64 (1927).
2. Discussion in Int. Units and Stds., *Brit. J. Radiol.* **30,** 285 (1925). (Nominating committee set up).
3. Int. X-ray Unit of Intensity, *Brit. J. Radiol.* **1,** 363 (1928).
4. *Radiology*, **22,** 289 (1934).
5. Roentgen Ray Standards and Units (Standardizing Procedure of the National Laboratories), *Amer. J. Roentgenol.* **31,** 815 (1934). (Also German, French, Swedish and British refs.).
6. *Radiology*, **23,** 580 (1934).
7. Roentgen Ray Standards and Units (Standardizing Procedure of the National Laboratories), *Amer. J. Roentgenol.* **31,** 815 (1934). (Also German, French, Swedish and British refs.).
8. *Amer. J. Roentgenol.* **39,** 295 (1938); *Radiology*, **29,** 634 (1937).
9. Report of ICRP and ICRU. NBS Handbook 47
10. *Amer. J. Roentgenol.* **71,** 139 (1954); *Radiology* **62,** 106 (1954).
11. Reports submitted to the International Commission on Radiological Units and Measurements, Copenhagen, July 1953, *Acta Radiol.* Suppl. 117 (1954).
12. *Report of the ICRU.* NBS Handbook 62 (1957).
13. *NBS Tech. News Bulletin* **41** 190 (1957).

MEMBERS PREPARING REPORTS OF THE ICRU

1956, Geneva

L. S. Taylor, Chairman (United States)
W. J. Oosterkamp, Secretary (Netherlands)
A. Allisy (France)
R. H. Chamberlain (United States)
F. Ellis (Great Britain)
G. Failla (United States)
L. H. Gray (Great Britain)
H. Holthusen (Germany)
H. E. Johns (Canada)
B. Rajewsky (Germany)
R. Sievert (Sweden)

1953, Copenhagen

H. M. Hansen, Hon. Chairman (Denmark)
W. V. Mayneord, Acting Chairman (Great Britain)
L. S. Taylor, Secretary (United States)
A. Allisy (France)
W. Binks (Great Britain)
R. H. Chamberlain (United States)
F. Ellis (Great Britain)
G. Failla (United States)
H. Holthusen (Germany)
H. E. Johns (Canada)
W. J. Oosterkamp (Holland)
B. Rajewsky (Germany)
R. Sievert (Sweden)

1950, London

W. V. Mayneord, Hon. Chairman (Great Britain)
L. S. Taylor, Secretary (United States)
W. Binks (Great Britain)
R. Coliez (France)
F. Ellis (Great Britain)
G. Failla (United States)
H. Holthusen (Germany)
G. C. Laurence (Canada)
R. R. Newell (United States)
W. J. Oosterkamp (Holland)
B. Rajewsky (Germany)
R. Sievert (Sweden)
A. Tivoli (Italy)

1937, Chicago

G. Failla, Hon. Chairman (United States)
I. Solomon, Chairman (France) Executive Sub-committee
L. S. Taylor, Secretary (United States) Executive Sub-committee
G. Schwarz (Austria)
J. Juul (Denmark)
H. M. Hansen (Denmark)
H. Holthusen (Germany)
H. Behnken (Germany)*
W. Friedrich (Germany)
F. L. Hopwood (Great Britain)*
G. W. C. Kaye (Great Britain)
N. S. Finzi (Great Britain)
A. Lambadarides (Greece)
D. den Hoed (Holland)
A. Bouwers (Holland)
M. Ponzio (Italy)
E. Pugno-Vanoni (Italy)*
M. Tanaka (Japan)
K. Inouye (Japan)
J. Jovin (Roumania)
E. C. Ernst (United States)
R. R. Newell (United States)

1934, Zurich

P. Scherrer, Hon. Chairman (Switzerland)
H. Koenig, Hon. Vice-chairman(Switzerland)
H. Holthusen, Hon. Secretary (Germany)
E. A. Owen, Hon. Secretary (Great Britain)*
H. Behnken (Germany)*
F. Běhounek (Czechoslovakia)
A. Bouwers (Holland)
E. C. Ernst (United States)
N. S. Finzi (Great Britain)
G. Guëben (Belgium)
L. Hass (Hungary)
H. M. Hansen (Denmark)
S. A. Heyerdahl (Norway)
D. den Hoed (Holland)
F. Holweck (France)
A. Liechti (Switzerland)
M. Ponzio (Italy)
E. Pugno-Vanoni (Italy)*
R. M. Sievert (Sweden)
I. Solomon (France) (Chairman, Exec. Subcommittee)
L. S. Taylor (United States) (Secretary, Exec. Subcommittee)
K. T. Watanabe (Japan)

---

* Members, Executive Subcommittee

January, 1957.

Health Physics Pergamon Press 1958. Vol. 1. pp. 315–331. Printed in Northern Ireland

ON THE RELATIONSHIP BETWEEN THE CHARGE OF AN ION AND ITS VELOCITY

J. NEUFELD

Health Physics Division, Oak Ridge National Laboratory, Operated by Union Carbide Corporation for the Atomic Energy Commission, Oak Ridge, Tennessee

(*Received* 6 *February* 1958)

Abstract—The Bohr–Lamb criterion is based on the assumption that an ion when passing through matter is stripped of all its orbital electrons that have velocity smaller than the translational velocity of the ion. This criterion is not based on any rigorous theoretical arguments, and its order of accuracy has never been determined, partly because it was correlated with statistical atomic models which represent merely a rough approximation.

In this investigation an attempt has been made to establish the possible accuracy of the Bohr–Lamb criterion by using, instead of a statistical model, the very reliable data on ionization potentials of various ions obtained by LISITZIN. The Lisitzin data are based on a study of various sequences of isoelectronic systems and provide values for electronic orbital velocities that are of a relatively high order of accuracy. These values have been used in order to correlate the Bohr–Lamb criterion with experimental data. The results show that Bohr–Lamb criterion is only applicable to fission fragments and not to lighter ions. In order to remedy this situation, the procedure suggested by KNIPP and TELLER was followed, and two empirical parameters, γ_L and α_L, were introduced. They are defined as follows:

$$\gamma_L = \frac{v_L}{V}, \quad \alpha_L = \frac{Z_L^{*(1)}}{Z_{\text{av}}^*},$$

where V and Z_{av}^* represent the velocity and the average charge of a moving ion, v_L is the velocity of the most loosely bound electron having a binding charge $Z_L^* = Z_{\text{av}}^*$, and Z_L^* is the binding charge that corresponds to the most loosely bound electron having velocity $v_L = V$. (The values v_L and $Z_L^{*(1)}$ have been determined from graphs based on the computations of LISITZIN.)

The numerical values of γ_L and α_L corresponding to all available measurements of ions of various velocities were determined. The results fail to show any regularity in the behavior of these parameters as a function either of the ionic species, or of the ionic velocity, or of the surrounding medium. After a survey of the literature covering the last 16 years was made, no definite opinion regarding the accuracy of the Bohr–Lamb criterion was reached. This may be due to the paucity of experimental data or to the lack of uniformity in the quality of various measurements made on ionic charges and velocities.

1. INTRODUCTION

(a) *General*

A HEAVY particle passing through matter undergoes very complicated processes of electron capture and loss as a result of which the charge of the particle continually fluctuates. The problem of determining the average charge of the particle has received considerable attention as a result of the pioneer investigations of BOHR[1] and LAMB[2] that followed the discovery of fission. Within recent years, owing to the rapidly increasing technological progress in the atomic energy field, this problem has become of importance in a great variety of fields such as in the study of damage in various substances due to fast fission products and other heavy particles, in biological studies, in the determination of absorbed dose, in various investigations dealing

with the fission process, etc. There is also a considerable interest in the development of efficient techniques for producing highly stripped ions and recent investigations relating to the effect of the medium on the charge of a moving ion would be of primary importance in that connection. There are now two projects in the United States dealing with accelerators specifically designed to accelerate highly stripped ions. One of these is located at the Radiation Laboratory of the University of California and the other at Yale University.

At the present time there appears to be no rigorous and usable theory that would serve to predict the charge of a known ionic species of known energy and moving in a known medium. A quantum mechanical approach has been successfully used in simple cases dealing with hydrogen atoms moving through a gas consisting of atomic hydrogen and within recent years a number of papers have appeared on this subject.[3–8] However, there is little likelihood that this approach will lead, at least in the immediate future, to successful practical results that would be applicable to heavier ions, such as fission fragments. Other methods have been used that are mainly based on statistical considerations and in many instances the results obtained are not in a satisfactory agreement with the experiment. Therefore, empirical factors have been applied in order to "patch up" the discrepancies between the theory and experiment.

In the present investigation we shall consider methods based on the Bohr–Lamb theory, on the Dmitriev theory, and methods involving studies of charge exchange collisions. These methods will be correlated with experiments in order to determine insofar as it is possible the accuracy of the proposed theories.

(b) *Bohr–Lamb theory*

This theory formulated independently by BOHR[1] and by LAMB[2] is based on an assumption that a "heavy" ion (i.e. an ion having a large nuclear charge Z) when passing through a medium is stripped of all its orbital electrons that have velocities smaller than the translational velocity of the ion. This assumption is based on essentially qualitative considerations. Thus BOHR argued that when a moving ion collides with the atoms of the surrounding medium the less firmly bound electrons are more easily removed, and when capture takes place it is more likely to occur in orbits associated with velocities of the same order of magnitude as the translational velocity of the ion. Somewhat different considerations have been presented by LAMB who argued that the moving ion is stripped of its outermost electrons until the ionization potential of the next stage of ionization is greater than the kinetic energy of electrons bombarding the ion with velocity V where V is the translational velocity of the ion.

Since the Bohr–Lamb theory is not supported by rigorous arguments it would be desirable to ascertain as to whether it is in agreement with experimental data and if such an agreement exists, what is its margin of accuracy. Such work was undertaken soon after the formulation of this criterion; however, due to the paucity of experiments to date, it is difficult to give a relatively complete evaluation of the merits of the criterion.

It appears from published literature that two models of ions have been used in order to correlate the experimental data with the electronic orbital velocities, i.e. the BOHR model[9] and the KNIPP and TELLER model.[10] Both models are statistical and give values of relatively low order of accuracy. Some experimenters correlated their measurements with these statistical models claiming a correlation of a higher order of accuracy than the one inherent in the model. It is somewhat surprising to see that the more refined calculations of electron binding energies in various ions have been apparently completely ignored. This applies particularly to the values given by LISITZIN[11] which have been known since 1938 and published in 1943 in a general reference manual.[12] Some of the estimates made by LISITZIN have been further refined by FINKELNBURG.[13,14] A complete table of the ionization potentials of all the electronic orbits in various ions has been computed by MAYER in collaboration with the Los Alamos computing group. These calculations are based on a Los Alamos report of MAYER.[15]

(c) *Dmitriev theory*

The theory formulated by DMITRIEV[16] is based on the following assumptions:

(1) Consider a monoenergetic beam of N atoms passing through a medium with velocity V and losing some of their orbital electrons as a result of the interaction with the medium. Let N_i designate the number of those atoms that have lost an electron belonging to the i^{th} orbit. The value $N_i/N = P_i$ is represented as:

$$P_i = P(V/v_i) \tag{1}$$

i.e. it is assumed that there is a unique dependence of the value P_i on the ratio between the translational velocity V of the ion and the orbital velocity v_i of the escaped electron. This dependence is applicable to any atom including hydrogen, and therefore the expression (1) corresponds to the ratio of $H^+/(H^0 + H^+)$ and represents the fraction of hydrogen beam that is positively charged as a result of its interaction with the surrounding medium. The function (1) has been determined by DMITRIEV from the measurements on hydrogen beams made by BARTELS,[17] KANNER,[18] and STIER *et al.*[19]

It should be noted that the event represented by the function P_i is likely to occur if i represents the external electronic orbit. On the other hand the situation in which an internal electronic orbit of the order i is removed from the atom while all the external orbits remain undisturbed is not likely to occur.

(2) The events represented by P_i for various values of i are assumed to be independent of each other and accordingly, the average charge of an ion has been represented as:

$$Z_{av}{}^* = \Sigma P_i \tag{2}$$

The above assumptions are somewhat controversial and a physical justification would be desirable. The values v_i in the expression (1) have not been explicitly defined, i.e. it has not been stated whether v_i represent orbital velocity in a neutral atom or in a partly ionized atom. However, it appears from the assumptions made by DMITRIEV that v_i should refer to a neutral atom.

2. CORRELATION OF THE BOHR–LAMB THEORY WITH EXPERIMENT

(a) *Model of an ion*

In order to ascertain the validity of the Bohr–Lamb theory, one needs to correlate the measurements of the ionic charges with a model of an ion in which the orbital velocity of the most external electron is known as a function of its binding charge. A model has been suggested by BOHR[9] which is a simplification of the Thomas–Fermi atom. Assuming that $n = 1$ represents the outermost electron and $n = 2, 3, 4$ represent electrons having correspondingly increasing binding energies, this model yields for the orbital velocity v_n of the n^{th} electron the following value:

$$v_n = \frac{n}{\nu} \text{ atomic units} \tag{3}$$

where ν is the "effective quantum number". (The expression (3) and all subsequent formulas shall be expressed in atomic units.[20]) For an outermost electron in a neutral atom $\nu \sim 1$. For inner electrons ν increases with n and reaches a value approximately equal to $Z^{1/3}$ for n somewhat smaller than $Z/2$ where Z is the atomic number. When n is increased further, ν decreases in value and reaches $\nu \sim 1$ for the most firmly bound electron. These estimates are very approximate and presumably can be applied to a neutral atom as well as to an ion in which a large portion of orbital electrons has been stripped off. This model has been subsequently modified by LINDHARD and SCHARFF[21] who used instead of $\nu \sim Z^{1/3}$ the value $\nu \sim KZ^{1/3}$ where $K < 1$.

According to BOHR the value $\nu \sim Z^{1/3}$ corresponding to intermediate values of n shows a flat maximum that is applicable to a relatively large number of orbital electrons and, therefore, he assumed that a charge of a fission fragment can be roughly represented as:

$$Z_{av}{}^* \sim Z^{1/3}V \tag{4}$$

The expression (4) agrees well with the experimental values for a heavy fission fragment ($Z = 54$) having $V = 4$ since it gives $Z^* = 15$. However, for a light fission fragment ($Z = 38$) having $V = 6$ we obtain from equation (4)

$Z^* = 20$ while the measured value is about 16. According to BOHR and LINDHARD,[22] the agreement for a heavy fission fragment is due to the fact that Z^* is somewhat smaller than $Z/2$ and therefore the value $\nu \sim Z^{1/3}$ is applicable whereas for a light fission fragment the value Z^* is not "somewhat smaller" than $Z/2$ and, therefore, a discrepancy occurs. Unfortunately the Bohr model is not defined more specifically for cases in which ν differs from $Z^{1/3}$ and therefore the practical applicability of this model is very limited.

A more detailed study of the statistical model of the Thomas–Fermi ion has been made by KNIPP and TELLER[10] and BRUNINGS *et al.*[23] These authors have plotted a series of graphs representing for each ion a functional dependence between the velocity of the most loosely bound electron v_{TF} and the corresponding binding charge Z^*_{TF}. The most loosely bound electron has been defined in case (a) as the outermost electron and in case (b) as the electron having the lowest binding energy. The authors have introduced an empirical parameter γ such that:

$$\gamma = \frac{v_{TF}}{V} \tag{5}$$

where v_{TF} is the velocity of the most loosely bound electron having a binding charge $Z_{TF}^* = Z_{av}^*$. Under the assumption (a), γ was found to decrease with atomic number ($\gamma = 0.6$ for $Z = 6$ to $\gamma = 0.35$ for $Z = 55$) and under the assumption (b) γ was found to increase with atomic number ($\gamma = 1.3$ for $Z = 6$ to $\gamma = 1.8$ for $Z = 55$). It should be kept in mind that no accurate measurements of ion charges were available at the time this investigation was made. For determining charges other than those of fission fragments the authors relied on range energy measurements in various gas mixtures and in order to determine the ion charges they used an approximate relationship:

$$\sigma_e = \langle i^2 \rangle_{av} \sigma_e' \tag{6}$$

where σ_e is the "electronic" component of the stopping power, $\langle i^2 \rangle_{av}$ is the mean square charge of the ion and σ_e' is the "specific electronic cross-section." Some of the errors resulting f m the above formula are due to the neglect of the influence of the medium and also to the fact that the contribution of the stopping power due to electron capture and loss processes has been neglected.

One of the fundamental merits of the treatment by BRUNINGS *et al.* is its potential usefulness for immediate practical purposes since by introducing a semi-empirical parameter such as γ the accuracy of the estimates involving the Bohr–Lamb criterion may be considerably improved. However, the parameter γ would be of value only if a sufficient number of experiments would establish its adaptability for practical purposes. It would also be desirable to correlate the experimental data not with a statistical ion but with a more refined model based on the calculations of LISITZIN, MAYER, and FINKELNBURG since these calculations take into account the individual behavior of each orbital electron.

The calculations of LISITZIN are based on empirical rules concerning a sequence of isoelectronic systems, i.e. systems having the same number of electrons and obtained by removing one, two, three . . . electrons from the first, second, and third . . . succeeding elements in the periodic table (such as, for instance, Na, Mg^+, Al^{++}, Si^{+++}, P^{++++}, etc.). The regularity of the ionization potentials in an isoelectronic sequence has been studied by many investigators in the past and a review of this subject together with a compilation of a great many experimental data is contained in LISITZIN's paper. Some of the estimates of LISITZIN have been further refined by FINKELNBURG on the basis of different semiempirical considerations.

The calculations made by MAYER utilize shielding constants similar to those of SLATER.[24] The ionization potential is then of the form:

$$I = \frac{(Z - S)^2}{2n^2} \tag{7}$$

where n is the principal quantum number of the outermost electron and the term S is due to the shielding by the other electrons in the ion. The shielding terms are obtained by means of hydrogenic wave functions as shown in MAYER's report which gives a table of their values.

The results of MAYER are not accurate for nearly neutral ions but they give very acceptable results for highly ionized ions. It is noted that semi-empirical considerations of LISITZIN and FINKELNBURG and theoretical considerations of MAYER lead to independent results that are in a quite good agreement one with another and also in good agreement with the experimental data.

(b) *Measurements*

From a survey of the literature it appears that many experiments were made on ions of hydrogen and helium and relatively few experiments dealt with heavier ions. The only ion species on which some experimental data are published are Li, N, O, F, Ne, Sr, and Xe. These data are plotted in Figs. 1–7 in which Z_{av}* represents the charge of an ion and V the corresponding velocity.

The data relating to Sr and Xe are meant to correspond to light and heavy fission fragments, respectively. Each of these fragments has a continuous spread in atomic number and both Sr and Xe are considered as "the most representative" species. The data relating to lighter ions are definitely associated with a specific ionic species. The measurements on fission fragments made originally by LASSEN[25,26] are indicated in Figs. 6 and 7 with full circles and the corresponding data recently obtained by FULMER and COHEN[27] are indicated with open circles. These two sets of measurements differ mainly in the determination of the velocity of fission. Thus, as indicated by LEACHMAN and SCHMITT,[28] the initial velocity of heavy and light fission fragments are $v = 4.42$ and $v = 6.48$ whereas the values given by LASSEN are $v = 4$ and $v = 6$, respectively. If we take into account the discrepancy in the velocity determination, we obtain a quite satisfactory agreement between the measurements of LASSEN and FULMER and COHEN in such media as hydrogen and helium (difference is less than 2 per cent). However, there is a discrepancy in the measurement in argon. Thus, for $v = 6.6$ for light fragments the value reported by LASSEN is $Z_{av} = 15.4$. Using the correspondence between the above sets of measurements we assume that the velocity of heavy and light fission fragments $v = 3$ and $v = 5$ as obtained by LASSEN correspond to velocities $v = 3.3$ and $v = 5.5$ in the measurements of FULMER and COHEN.

In order to correlate the above experimental data with values that could be obtained by means of the Bohr–Lamb theory, we have plotted in Figs. 1–7 curves representing the relationship between the outermost orbital velocity v of an ion and the corresponding binding charge Z*. The curves designated as L, M, and F are based, respectively, on the estimates of LISITZIN, MAYER, and FINKELNBURG. The electron velocities are calculated assuming that each electron is in a central coulomb field characterized by the corresponding binding charge. These curves represent an interpolation of the data of LISITZIN, MAYER, and FINKELNBURG in which the discrete points representing the binding energies and velocities of neighboring orbits have been connected by straight line segments. Thus each of these curves represents the relationship between the velocity of the ion and the corresponding charge when the following two assumptions are satisfied: (a) a stripping of the outermost orbit occurs only when the velocity V of the ion exceeds by a very small amount the velocity v_n of the orbital electron and as a result of the stripping the charge of the ion is equal to the binding charge corresponding to the orbital electron of velocity v_n; (b) the charge of the ion Z* is a linear function of its velocity v over a short range bounded by the velocities of two neighboring electron orbits.

It is apparent from Figs. 1–7 that the Bohr–Lindhard theory can give, at best, a rough estimate of the average charge of the moving ion and, therefore, in order to obtain more refined values, we have to consider the applicability of an empirical parameter. Furthermore, it is seen that the average charge is not a unique function of the velocity, and therefore, we have to provide such an empirical parameter that would take into account the nature of the stopping medium. We shall consider the dependence between the charge of a moving ion and the following parameters: the species of the stopping atom, the pressure of the

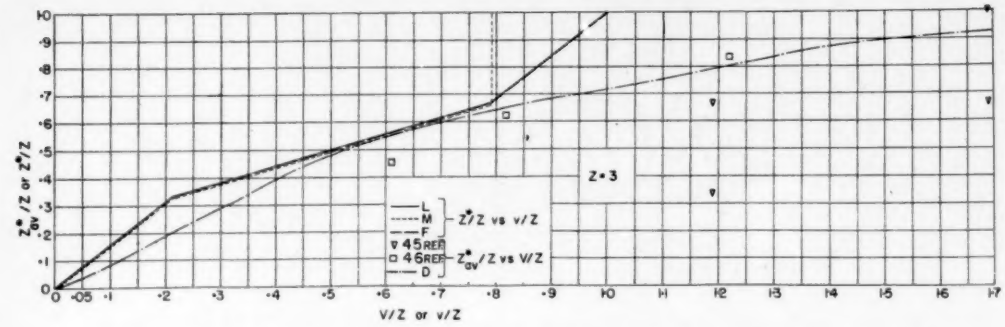


FIG. 1. Electronic orbital velocities and average charge for lithium ions ($Z = 3$). References as indicated in Reference List,

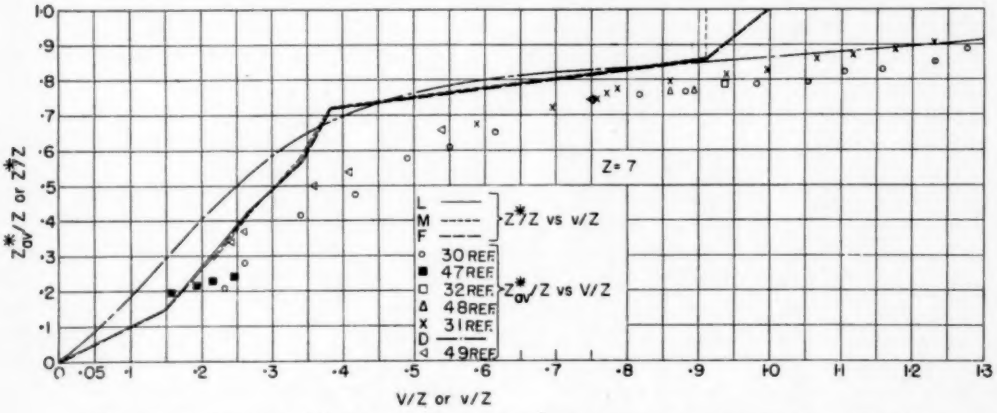


FIG. 2. Electronic orbital velocities and average charge for nitrogen ions ($Z = 7$). References as indicated in Reference List.

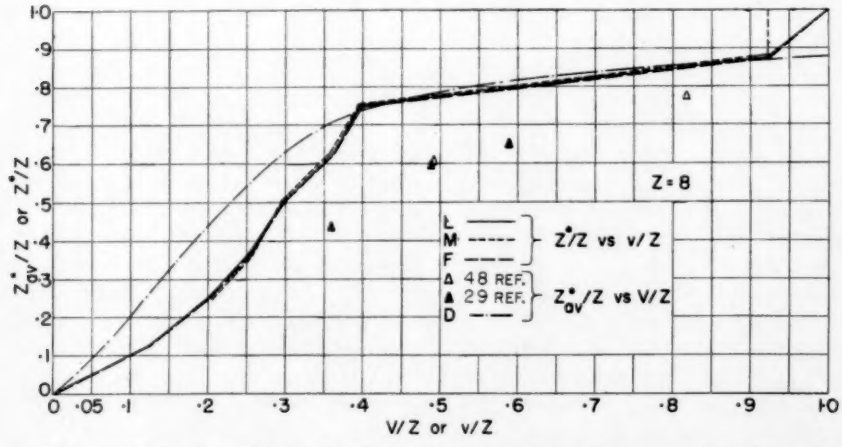


FIG. 3. Electronic orbital velocities and average charge for oxygen ions ($Z = 8$). References as indicated in Reference List.

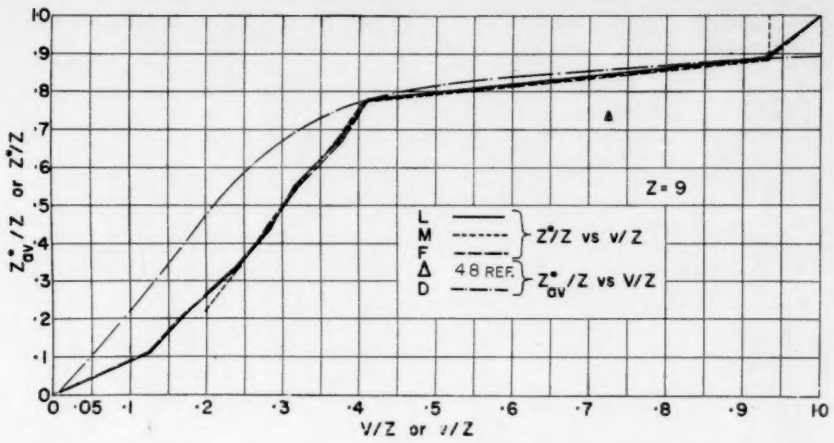


FIG. 4. Electronic orbital velocities and average charge for fluorine ions ($Z = 9$). References as indicated in Reference List.

medium if the particle passes through a gas, and the effect of the state of condensation of the medium.

(1) *The species of the stopping atom*

(a) *Gaseous medium.* According to LASSEN, Z_{av}* of a "heavy ion" in a gaseous medium depends very significantly on the species of the stopping atom. Thus a heavy fission fragment at $V = 4$ has Z_{av}* about 25.8 per cent higher in argon than in helium. HUBBARD and LAUER[29] performed measurements similar to those made by LASSEN on neon ions having velocity $V = 3.77$ in hydrogen, helium, and nitrogen, and on oxygen ions having velocity $V = 4.25$ in nitrogen, krypton, and xenon and reported no significant dependence between the average charge of the ion and the nature of the medium. However, they did not preclude the possibility of differences in magnitude of the order of 10 per cent.

(b) *Solid or liquid medium.* It is of interest to note that the dependence between Z_{av}* and the character of the stopping atom is

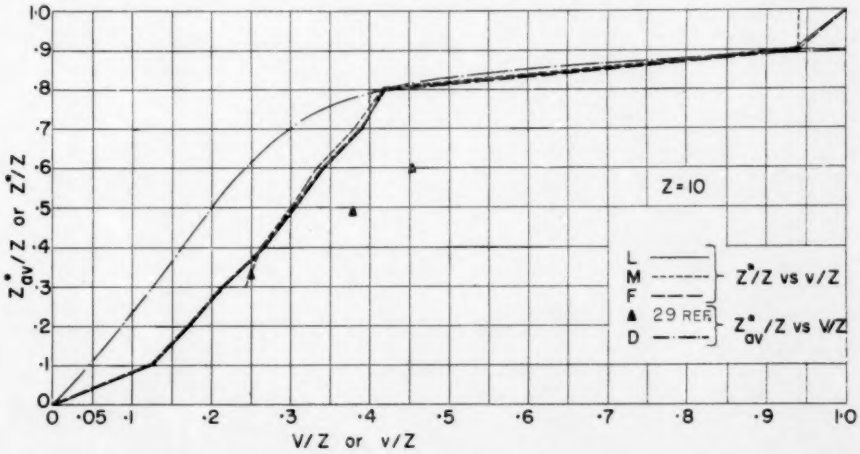


FIG. 5. Electronic orbital velocities and average charge for neon ions ($Z = 10$). References as indicated in Reference List.

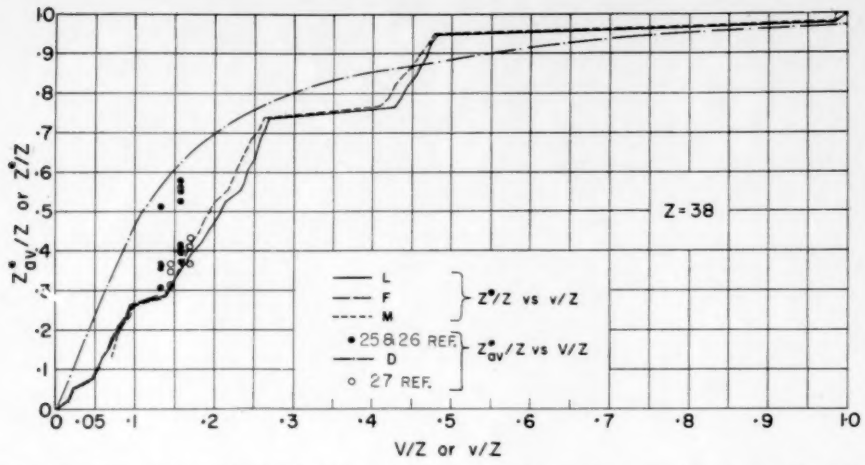


FIG. 6. Electronic orbital velocities and average charge for strontium ions ($Z = 38$). References as indicated in Reference List.

somewhat less pronounced in a condensed medium than in a gas. According to LASSEN the variation in values of Z_{av}* for fission fragments passing through various solids is less than 10 per cent. A similar dependence can also be seen by comparing the measurements of Z_{av}* measured by REYNOLDS *et al.*[30] in nickel ($Z = 28$) with the corresponding values measured by REYNOLDS *et al.*[31] in Formvar ($Z \sim 4$) and a comparison of these two measurements shows rather unexpectedly that Z_{av}* is higher in a stopping medium having a lower atomic number. A similar result has been obtained by LASSEN who found that the charge of a fission fragment decreases if the atomic number of the stopping medium is higher.

(2) *The pressure of the gaseous medium*

According to LASSEN[26] the average charge of a fission fragment increases with the pressure. Thus the Z_{av}* for a fission fragment in argon at a pressure of a few millimeters of Hg is about 16. This charge increases relatively

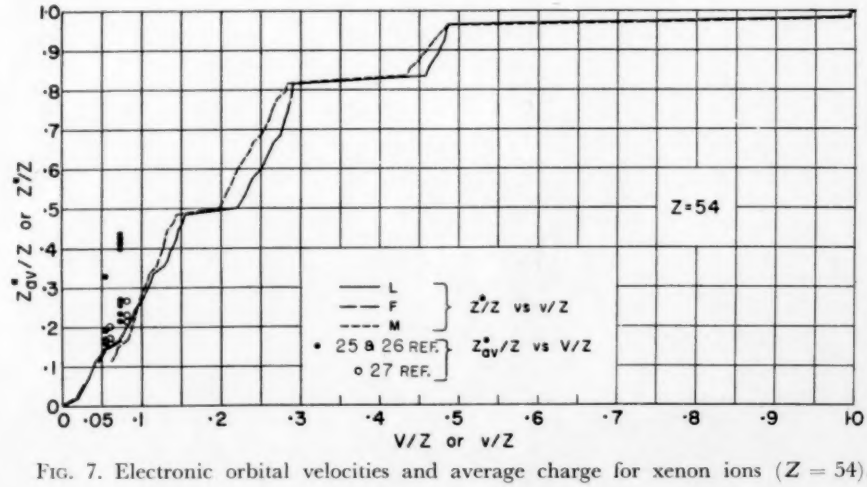


FIG. 7. Electronic orbital velocities and average charge for xenon ions ($Z = 54$).

fast with the pressure until it reaches a limiting value of 19 at a pressure of about 70 mm of Hg. For pressures exceeding this value, it increases very slowly. This dependence between the pressure of the gas and the charge of the ion has not been observed by HUBBARD and LAUER in their experiments on ions of oxygen and neon. This is apparently due to the fact that the highest pressure that they used was about 0.1 mm of Hg and at this pressure the effect was not detectable.

The pressure effect, according to LASSEN, depends upon the relative values of two time intervals designated as T_{coll} and T_{exc}. The value T_{coll} represents the average time interval separating two successive charge exchange collisions between the moving ion and the stopping atoms, and the value T_{exc} is the radiative life time of the ion in an excited state. It is apparent that after a collision one or more orbital electrons carried by the ion are often left in an excited state. If the pressure is relatively high, we have $T_{\text{exc}} > T_{\text{coll}}$ and there is a high probability that an electron in an excited state will be removed from the ion in the next collision. The higher the pressure, the more frequent are the collisions in which the excited electron is removed from the ion and the more efficient is the process of stripping the external orbital electrons from the ion. On the other hand, for lower pressures we have $T_{\text{exc}} < T_{\text{coll}}$ and an electron that is initially excited as a result of a collision has time to return to the ground state before the next collision takes place. At the next collision the probability of knocking out the electron from the ground state is smaller than if the electrons were excited, and therefore, the process of stripping the ion from its orbital electrons is less efficient at a lower pressure.

Therefore, in order to have an appropriate ionic model which would take into account the pressure effect, one has to consider the distribution of electronic orbital velocities in an ion that is continually perturbed by collisions with the atoms of the stopping medium. Such models have not been submitted to date and both statistical models given by BOHR and by BRUNINGS *et al.* relate to an ion in the ground state.

The experimental data for $Z_{\text{av}}{}^*$ in gases plotted in Figs. 1–7 relate to measurements at very low pressures at which the pressure effect is not significant. In particular, the values for fission fragments are extrapolated to zero pressure. Thus a heavy fission fragment emitted from uranium into a vacuum ($p \sim 10^{-3}$ mm) in the deflection chamber has an average charge 21.6 resulting from the equilibrium attained within the mass of uranium. This charge decreases if the deflection chamber is filled with argon and as the pressure is increased. This charge reaches a minimum value of about 15 when the pressure $p \sim 0.5$ mm of Hg. Then when the pressure increases further, the charge increases linearly at a relatively small rate. LASSEN has extrapolated the linear part of this relationship to the value $p = 0$ and thus defined the equilibrium charge for zero pressure.

(3) *The state of condensation of the medium*

It has been observed that an ion passing through a solid or liquid has a higher charge than an ion of the same velocity passing through a gas. This effect may be considerable, particularly in case of heavy ions. Thus, according to LASSEN,[26] a fission fragment passing through helium at velocity $V = 3$ has an average charge 8.6 but the same ion emerging with the same velocity from a layer of mica has a charge 109 per cent higher. Unfortunately the experimental evidence regarding the state of condensation effect for lighter ions is not very conclusive. According to HUBBARD and LAUER[29] an oxygen ion at $E = 6.1$ MeV has a charge $Z_{\text{av}}{}^* = 4.8$ in a gas and according to STEVENS and WALKER,[32] the ion at $E = 6.2$ MeV has a charge $Z_{\text{av}}{}^* = 4.84$ in a solid. These two measurements would indicate that the state of condensation effect is not significant. On the other hand, the measurements of ALLISON, CASSON, and WEYL[33] suggest that the dependence on the medium is very noticeable for ions of helium. According to ALLISON (private communication) the ratios of He^{++}/He^{+} for metals as measured by ALLISON, CASSON, and WEYL are substantially higher than the corresponding values for air measured by SNITZER[34] and the ratios

of He^{++}/He^{+} in metals measured by DISSANAIKE[35] are substantially higher than the values obtained in gases by SNITZER and by STIER *et al.*[19] There is, however, some discrepancy between the values He^{++}/He^{+} in metals as obtained by DISSANAIKE and the corresponding values obtained by ALLISON, CASSON, and WEYL.

In order to account for the state of condensation effect, BOHR and LINDHARD[22] suggested an explanation somewhat similar to the one given in connection with the "pressure effect". They assume that immediately after a collision a heavy ion has many of its orbital electrons in excited states and that the excitation energy of the electrons is not uniformly distributed. Some of the electrons have a relatively large excitation energy and it would require little additional energy in order to leave the ion. However, these excited electrons tend to readjust themselves so as to distribute their excitation energy more uniformly. Let T_{dis} be the time necessary for such a redistribution. The occurrence of the state of condensation effect depends upon the relative values of T_{dis} and T_{coll} in the same manner as the pressure effect depends upon the relative values of T_{exc} and T_{coll}.

For gases we have $T_{\text{coll}} > T_{\text{dis}}$ and, therefore, the excited ion distributed its excitation energy among several electrons before the next collision and the chance of having the ion further ionized is relatively small. For solids and liquids we have $T_{\text{coll}} < T_{\text{dis}}$ and, therefore, there is not sufficient time between two collisions to redistribute excitation energy and an electron originally placed in an excited state as a result of the first collision has a relatively large probability of being removed as a result of the next collision.

In order to apply the Bohr–Lamb theory to a condensed medium it would be helpful to know the distribution of electronic orbital velocities in an ion that is continually perturbed by collisions at a relatively high frequency. Such a study has not been made; therefore, another alternative would be to consider an empirical factor. Thus, BOHR and LINDHARD applied an empirical factor equal to 3/2, i.e. they assumed the average charge of a fission fragment in a solid is roughly 3/2 times larger than the corresponding charge in a gas. This factor, however, has a very limited region of applicability.

There is a question as to whether the model based on the BOHR and LINDHARD theory is sufficient to take account of the state of condensation effect. According to NEUFELD[36] the state of condensation effect may be due to a different process, i.e. to the reaction of the medium perturbed by the moving ion. This reaction causes a polarizing field and results in spontaneous emission of the orbital electrons. It is noted that NEUFELD's explanation is applicable to any ion whereas the BOHR and LINDHARD explanation is applicable only to heavy ions.

According to a recent suggestion of NEUFELD and SNYDER[36] the state of condensation effect in heavy ions has two contributing factors. One of these is due to the "close collisions" between the moving ions and the stopping atoms, and it accounts for a portion of this effect that is based on the BOHR and LINDHARD explanation. The other factor is due to "distant collisions" and it represents mainly the integrated action of the medium on the moving ion. This other factor represents the portion of the effect that is based on the explanation of NEUFELD. Therefore, in order to apply directly the Bohr–Lamb theory and to take account of the state of condensation effect one would need a model of an atom that is periodically perturbed and polarized in a strong electrical field. Such a model does not exist. Therefore, we have to follow the general procedure outlined by KNIPP and TELLER which is based on a model of an ion in the ground state free of any external disturbance and see whether we can correct our results in order to account the effect of the medium by properly modifying the empirical parameter disturbance and to correct any discrepancy due to this model by means of an appropriate empirical parameter.

(c) *Empirical parameters*

We have introduced two parameters designated as γ_L and α_L. The parameter γ_L is

similar to the parameter γ used by KNIPP and TELLER and by BRUNING *et al.*, i.e.

$$\gamma_L = \frac{v_L}{V} \tag{8}$$

where v_L is the velocity of the most loosely bound electron having a binding charge $Z_L^* = Z_{\text{av}}^*$. The parameter α_L is such that:

$$\alpha_L = \frac{Z_L^{*(1)}}{Z_{\text{av}}^*} \tag{9}$$

where $Z_L^{*(1)}$ is the binding charge that corresponds to the most loosely bound electron having velocity $v_L = V$. The subscript L indicates that the above values relate to the "Lisitzin graphs".

The values γ_L and α_L are shown in Table 1 for gaseous media and in Table 2 for condensed media. It is difficult to establish any regularity in the behavior of either of these parameters. For many ion species the parameter γ_L or α_L seems to decrease with the increase of the velocity of the ion. However, this behavior is not sufficiently general.

Table 3 gives some general data regarding γ_L and α_L such as the maximum and minimum values for gases and condensed media, respectively In this table we differentiate between "fission fragments" corresponding to $Z = 38$, 54, and "lighter ions" corresponding to $Z = 3$, 7, 8, 9, 10.

3. CORRELATION OF THE DMITRIEV THEORY WITH EXPERIMENT

The function $P(V/v_i)$ has been determined from the measurements of STIER *et al.*,[19] STIER and BARNETT,[37] and BARNETT.[38] These measurements are plotted in Fig. 8 showing $P(V/v_0)$ (where $v_0 = e^2/\hbar$) for hydrogen beam moving in various gaseous media such as hydrogen, helium, oxygen, nitrogen, neon, and argon. We have neglected the occurrence of negative ions which are somewhat more numerous in argon for energies below 8 keV but are quite negligible for other gases. In the latter case the fraction of the total beam represented by the negatively charged ions is less than 2 per cent We have chosen for our computations the values $P(V/v_0)$ for the hydrogen beam in nitrogen since they represent the "average behavior". It is noted, however, that some of the curves shown in Fig. 8 depart considerably from the "average". The values v_i representing the orbital electronic velocities of neutral atoms have been taken from tables given by SLATER.[39] The dependence between the Z_{av}^* and the velocity V as predicted by the Dmitriev theory is plotted in form of graphs "D" in Figs. 1–6. It is noted that these graphs give values higher than those measured.

4. THEORIES BASED ON CHARGE EXCHANGE COLLISIONS AND THEIR CORRELATION WITH EXPERIMENT

BELL[40] has estimated the charge of a moving ion by considering more in detail the capture and loss processes due to the interaction with the surrounding medium. These processes can be described in the framework of classical mechanics provided no particular refinements are taken into account. By applying the statistical Thomas–Fermi model to heavy and light fission fragments, BELL has determined the functions $\sigma_L(Z^*, V)$ and $\sigma_C(Z^*, V)$ representing the loss and capture cross-sections (for one electron) corresponding to a fission fragment having charge Z^* and velocity V. Neglecting the possibility of transferring two or more electrons in a single encounter, he assumed that the most probable and presumably the average ionic charge Z_{av}^* for a given velocity V can be determined from:

$$\sigma_C(Z_{\text{av}}^*, V) \sim \sigma_L(Z_{\text{av}}^*, V) \tag{10}$$

The average charge of a fission fragment based on equation (10) has been computed by BELL and his values are shown in Table 1. Table 1 shows also for comparison the experimental values obtained by LASSEN[25,26] and the values of LASSEN modified by BELL who took into account that his values of momenta of fission fragments exceed those of LASSEN by 6 per cent.

According to GLUCKSTERN,[41] the method used by BELL for calculating $\sigma_C(Z^*, V)$ should be modified in order to represent the capture cross-section for any electron and not the sum of individual capture cross-sections as assumed by BELL. This modification will

Table 1. *Parameters* γ_L *and* α_L *for ions passing through gases*

Ion species	Stopping medium	Reference	V/Z	γ_L	α_L
Oxygen	Argon	29	0.36	0.77	1.42
			0.49	0.71	1.29
			0.58	0.63	1.22
Neon	Argon	29	0.25	0.92	1.10
	(hydrogen or helium or nitrogen or argon)		0.38	0.80	1.38
	Argon		0.45	0.75	1.35
Strontium	Hydrogen	24, 44	0.132	1.21	0.80
	Helium			1.09	0.92
	Nitrogen or argon			1.23	0.78
	Hydrogen	27	0.145	1.07	0.90
	Helium			1.00	1.00
	Argon			1.12	0.85
	Hydrogen	25, 44	0.158	1.16	0.84
	Helium			1.04	0.95
	Nitrogen			1.09	0.89
	Argon			1.12	0.87
	Oxygen			1.11	0.88
	Hydrogen	27	0.170	1.04	0.96
	Helium			0.95	1.07
	Argon			1.09	0.91
Xenon	Hydrogen	25, 44	0.055	1.35	0.88
	Helium			1.20	0.93
	Nitrogen			1.45	0.76
	Argon			1.44	0.77
	Hydrogen	27	0.061	1.18	1.15
	Helium			1.03	1.03
	Argon			1.31	0.77
	Hydrogen	25, 44	0.074	1.22	0.73
	Helium			1.14	0.79
	Nitrogen			1.30	0.67
	Argon			1.37	0.63
	Oxygen			1.32	0.66
	Hydrogen	27	0.082	1.08	0.91
	Helium			1.03	0.97
	Argon			1.22	0.79

Table 2. Parameters γ_L and α_L for ions passing through solids

Ion species	Stopping medium	Reference	V/Z	γ_L	α_L
Lithium	Photographic emulsion	45	1.19	0.66	
			1.69	0.59	
	Celluloid	46	0.61	0.67	1.26
			0.84	0.84	1.15
			1.22	0.23	
			1.83	0.52	
Nitrogen	Nickel	30	0.23	0.75	1.66
			0.26	0.78	1.49
			0.34	0.76	1.37
			0.42	0.69	1.54
			0.49	0.70	1.29
			0.55	0.64	1.25
			0.61	0.59	1.19
			0.75	0.63	1.09
			0.82	0.64	1.10
			0.88	0.64	1.11
			0.98	0.66	1.24
			1.05	0.63	
			1.10	0.71	
			1.16	0.69	
			1.23	0.72	
			1.28	0.73	
	Formvar	31	0.59		
			0.70		
			0.76		
			0.77		
			0.79		
			0.86		
			0.94		
			1.99		
			1.07		
			1.12		
			1.18		
			1.23		
	Nickel	47	0.16	1.06	0.88
			0.19	0.90	1.17
			0.21	0.85	1.33
			0.24	0.78	1.54
	Organic substance	48	0.86	0.65	1.11
			0.89		
		32	0.93	0.68	1.16

Table 2 (*continued*)

Ion species	Stopping medium	Reference	V/Z	γ_L	α^L
Nitrogen	Celluloid	49	0.24	0.97	1.06
			0.26	0.93	1.11
			0.36	0.84	1.28
			0.41	0.80	1.33
			0.54	0.68	1.16
			0.75	0.62	1.09
Oxygen	Organic substance	48	0.49	0.72	1.27
			0.82	0.61	1.09
Fluorine	Organic substance	48	0.72	0.55	1.15
Strontium	Mica	25, 44	0.132	1.61	0.55
	Beryllium		0.158	1.51	0.61
	Aluminum			1.49	0.63
	Silver			1.48	0.64
	Uranium			1.36	0.67
Xenon	Mica	25, 44	0.055	2.09	0.45
	Beryllium		0.074	1.96	0.39
	Aluminum			1.94	0.40
	Silver			1.90	0.41
	Uranium			1.86	0.43

Table 3. *Dependence between the parameters* γ_L *and* α_L *and the character of the medium*

		Gases		Condensed media	
		γ_L	α_L	γ_L	α_L
Fission fragments	Maximum value	1.45	1.15	2.09	0.67
	Minimum value	1.03	0.63	1.36	0.39
Lighter ions	Maximum value	0.92	1.42	1.06	1.66
	Minimum value	0.63	1.10	0.52	0.88

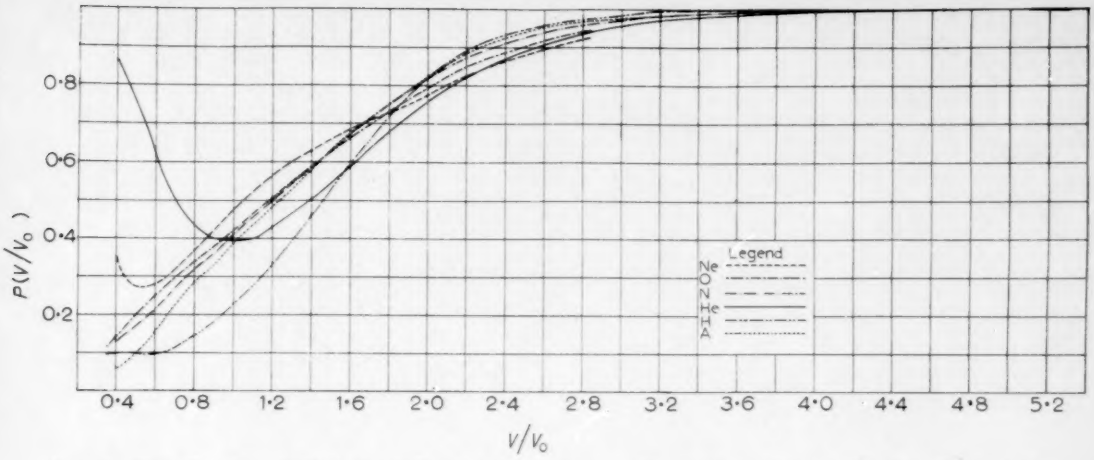


FIG. 8. The function $P(V/v_0)$ for hydrogen in various media such as Ne, O, N, He, H, and A.

cause a decrease by about 40 per cent of the value of the cross-section calculated by BELL. GLUCKSTERN questioned also the adequacy of a statistical model to describe the high charge states of a fission fragment and he modified BELL's method in order to make it particularly applicable to "light" ions (i.e. to ions having a relatively small nuclear charge Z). By replacing the Thomas–Fermi distribution by one in which the ion electrons are located in concentric shells with radii chosen to correspond to the known ionization potentials, GLUCKSTERN determined the charge distribution for the ions of oxygen, neon, and nitrogen in gases by means of the expressions:

$$N(Z^*, V)\,\sigma_L(Z^*, V) = N(Z^* + 1, V)\,\sigma_C(Z^* + 1, V)$$

$$\sum_{Z^*} N(Z^*, V) = 1 \qquad (11)$$

in which $N(Z^*, V)$ is the probability that the ion will have a charge Z^* at velocity V. The values of the average charge obtained by GLUCKSTERN for oxygen and neon are in relatively good agreement with the experimental data of HUBBARD and LAUER,[29] the largest discrepancy being 0.3 of a unit charge.

Table 4. Charges of fission fragments obtained by BELL *and by* LASSEN

Stopping medium	Ion species	Measurements of LASSEN	Modified values of LASSEN	Values calculated by BELL
Hydrogen	Light fragment	15.8	16.7	22.5 (21)
	Heavy fragment	12.6	13.3	16.3 (15)
Helium	Light fragment	14.1	14.9	16.5
	Heavy fragment	11.6	12.2	12.5
Oxygen	Light fragment	15.2	16.1 ± 0.5	15.5
	Heavy fragment	14.1	14.8 ± 0.5	13.5

The values obtained by GLUCKSTERN for nitrogen ions in gaseous media are somewhat smaller (by amounts varying from 0.2 to 0.4 of a unit charge) than the corresponding values for nitrogen ions that have been charge equilibrated in solids and measured by REYNOLDS *et al.*,[30] REYNOLDS and ZUCKER,[42] and REYNOLDS *et al.*[43] According to GLUCKSTERN, the model used in his computation is very crude and this discrepancy relating to the values of nitrogen ions should not be considered as a verification of the effect of the medium on the charge of the ion.

The expression (10) has also been used by BOHR and LINDHARD[22] in estimating the charge of fission fragments and the agreement with the experiment was found satisfactory. However, the estimates of cross-section for loss or capture are somewhat cruder than those of BELL.

The above methods based on the estimates of capture and loss cross-section appear to have a better claim to accuracy than those involving the Bohr–Lamb criterion. However, there are no measurements of cross-section for loss and capture covering any extended range of Z^* or V that could be used for comparison with the data obtained by BELL, GLUCKSTERN, or BOHR and LINDHARD.

FINAL REMARKS

Owing to the inaccuracies of atomic models the existing theories lead to very approximate results. By the use of data of LISITZIN, MAYER, and FINKELNBURG, the present paper provides a better basis for the evaluation of the accuracy of these theories.

A particularly interesting situation has arisen in connection with the possible accuracy of the Bohr–Lamb criterion. In order to apply this criterion to a great variety of cases, BRUNINGS, KNIPP, and TELLER have found it necessary to introduce an empirical parameter γ. This parameter has been applied in conjunction with a statistical model, and somewhat unexpectedly in spite of the low accuracy of the model, the experimenters report a satisfactory agreement with the measurements. In this investigation we introduced an ionic model of higher order of accuracy than the Thomas–Fermi model and by means of similar empirical parameters α_L and γ_L an effort has been made to obtain correlation with experimental data. It is found that such a correlation is not very satisfactory. We have reached, therefore, a somewhat paradoxical conclusion that an inexact model corrected by a semi-empirical parameter gives better agreement with the experiment than does a more nearly exact model similarly corrected.

In order to assist further in this field we have prepared and assembled a set of graphs showing the LISITZIN and MAYER curves and data of FINKELNBURG for those elements that have not been included in Figs. 1–7. This set of graphs is contained in a report issued by the Oak Ridge National Laboratory (Supplement to ORNL-2365, 1957).

Acknowledgments—The writer is indebted to Dr. HARRIS MAYER for unpublished data on ionization potentials of various ions, and to M. R. FORD for numerical calculations.

REFERENCES

1. N. BOHR, *Phys. Rev.* **58,** 654 (1940).
2. W. E. LAMB, *Phys. Rev.* **58,** 696 (1940).
3. D. R. BATES and A. DALGARNO, *Proc. Phys. Soc.* A **65,** 919 (1952).
4. D. R. BATES and G. W. GRIFFING, *Proc. Phys. Soc.* A **66,** 961 (1953).
5. D. R. BATES and A. DALGARNO, *Proc. Phys. Soc.* A **66,** 972 (1953).
6. D. R. BATES and G. W. GRIFFING, *Proc. Phys. Soc.* A **68,** 90 (1955).
7. A. DALGARNO and H. N. YADAV, *Proc. Phys. Soc.* A **66,** 173 (1953).
8. J. D. JACKSON and H. SCHIFF, *Phys. Rev.* **89,** 359 (1953).
9. N. BOHR, *Dan. Mat. Fys. Medd.* **18,** 8 (1948).
10. J. KNIPP and E. TELLER, *Phys. Rev.* **59,** 659 (1941).
11. E. LISITZIN, *Comm. Phys. Math. Soc. Sci. Fenn.* **10,** 4 (1938).
12. J. D'ANS and E. LAX, *Taschenbuch fur Chemiker und Physiker.* Springer, Berlin (1943).
13. W. FINKELNBURG, *Z. Naturf.* **2a,** 16 (1947).
14. W. FINKELNBURG and W. HUMBACH, *Naturwissenschaften* **42,** 35 (1955).
15. H. MAYER, *Method of Opacity Calculations,* Los Alamos Scientific Laboratory (1947) (AECD-1870).

16. I. S. Dmitriev, *Z. Eksperim i Teoret. Fiziki* **32,** 570 (1957).
17. H. Bartels, *Ann. Phys.* **13,** 373 (1932).
18. H. Kanner, *Phys. Rev.* **84,** 1211 (1951).
19. P. M. Stier, C. F. Barnett and G. E. Evans, *Phys. Rev.* **96,** 973 (1954).
20. D. R. Hartree, *Proc. Camb. Phil. Soc.* **24,** 89 (1928).
21. J. Lindhard and M. Scharff, *Dan. Mat. Fys. Medd.* **27,** 15 (1953).
22. N. Bohr and J. Lindhard, *Dan. Mat. Fys. Medd.* **28,** 7 (1954).
23. J. H. M. Brunings, J. K. Knipp and E. Teller, *Phys. Rev.* **60,** 657 (1941).
24. J. C. Slater, *Phys. Rev.* **36,** 57 (1930).
25. N. O. Lassen, *Dan. Mat. Fys. Medd.* **26,** 5 (1951).
26. N. O. Lassen, *Dan. Mat. Fys. Medd.* **26,** 12 (1951).
27. C. B. Fulmer and B. L. Cohen, In press.
28. R. B. Leachman and H. W. Schmitt, *Phys. Rev.* **96,** 1366 (1954).
29. E. L. Hubbard and E. J. Lauer, *Phys. Rev.* **98,** 1814 (1955).
30. H. L. Reynolds, D. W. Scott and A. Zucker, *Phys. Rev.* **95,** 671 (1954).
31. H. L. Reynolds, L. D. Wyly and A. Zucker, *Phys. Rev.* **98,** 474 (1955).
32. K. G. Stevens and D. Walker, *Phil. Mag.* **45,** 543 (1954).
33. S. K. Allison and S. D. Warshaw, *Rev. Mod. Phys.* **25,** 779 (1953).
34. E. Snitzer, *Phys. Rev.* **89,** 1237 (1953).
35. G. A. Dissanaike, *Phil. Mag.* **44,** 1051 (1953).
36. J. Neufeld and W. S. Snyder, *Phys. Rev.* **107,** 96 (1957).
37. P. M. Stier and C. F. Barnett, *Phys. Rev.* **103,** 896 (1956).
38. C. F. Barnett and H. K. Reynolds, In press.
39. J. C. Slater, *Phys. Rev.* **98,** 1037 (1955).
40. G. I. Bell, *Phys. Rev.* **90,** 548 (1953).
41. R. L. Gluckstern, *Phys. Rev.* **98,** 1817 (1955).
42. H. L. Reynolds and A. Zucker, *Phys. Rev.* **95,** 1353 (1954).
43. H. L. Reynolds, L. D. Wyly and A. Zucker, *Phys. Rev.* **98,** 1825 (1955).
44. N. O. Lassen, *Dan. Mat. Fys. Medd.* **30,** 8 (1955).
45. M. I. Kuznetzov, P. I. Lukirski and N. A. Perfilov, *Dokl. Akad. Nauk. SSSR* **100,** 665 (1955).
46. A. A. Teplova, I. S. Dmitriev, V. S. Nikolaev and L. N. Fateeva, *Zh. Eksperim i Teoret. Fiziki* **32,** 974 (1957).
47. M. I. Korsunski, L. I. Pivovar, A. M. Markus and K. L. Leviant, *Dokl. Akad. Nauk SSSR* **103,** 399 (1955).
48. K. G. Stevens and D. Walker, *Proc. Roy. Soc.* **229,** 376 (1955).
49. V. S. Nikolaev, L. N. Fateeva, I. S. Dmitriev and A. A. Teplova, *Zh. Eksperim i Teoret. Fiziki* **32,** 965 (1957).

Health Physics Pergamon Press 1958. Vol. 1. pp. 332–339. Printed in Northern Ireland

NEUTRON BEAM CHARACTERISTICS FROM THE UNIVERSITY OF CALIFORNIA 60 IN. CYCLOTRON

E. TOCHILIN* and G. D. KOHLER†
U.S. Naval Radiological Defense Laboratory, San Francisco 24, California

(*Received* 30 *June* 1958)

Abstract—Physical measurements of the neutron beams obtained by bombarding a thick Be target with 24 MeV deuterons and 12 MeV protons included a determination of fast neutron flux with sulfur threshold detectors and of spectrum with nuclear track plates. On the basis of these measurements the first collision neutron dose was computed. The γ-ray dose associated with the neutron beam was determined by means of photographic emulsion techniques. Comparisons were made with other calibrated neutron sources of similar spectra.

INTRODUCTION

DURING the past few years neutrons from the University of California 60 in. cyclotron at Crocker Laboratory have been used for a variety of biological investigations.[1,2] The cyclotron has served as an excellent instrument for this purpose. Neutron beams of high intensity are readily available which allow uniform irradiation of large groups of animals over a reasonable period of time. Aside from the initial problems of physical dosimetry, the neutron irradiation program has proceeded in a straightforward manner that differs little from the program involving the more conventional type of exposures made with X-rays.

The neutron spectra available from the cyclotron are determined by the energy of the different particles that may be accelerated and by the target material. Initial work on the cyclotron was carried out using neutrons produced by bombarding a thick Be target with 20 MeV deuterons. The resultant neutrons were emitted as a continuous spectrum with a maximum energy of 24 MeV and a population maximum at approximately 8 MeV.[3] While these investigations were in progress the cyclotron was modified and the energy of the deuteron beam was increased to 24 MeV. Protons accelerated by the cyclotron have half the energy of the deuteron beam. By use of a thick Be target and 12 MeV protons it was possible to obtain neutrons of considerably lower energy with a resultant spectrum similar to that of fission neutrons.

DOSIMETRY

Two techniques are available for evaluating neutron energy absorption in tissue: (1) direct measurement of energy loss with tissue-equivalent ionization chambers or proportional counters and (2) an indirect approach involving measurements of neutron flux density and energy distribution.

Earliest attempts to use ionization chamber techniques for neutron measurements were made by AEBERSOLD and ANSLOW,[4] employing a Victoreen 100 r thimble chamber. A tissue-equivalent ionization chamber which allowed a direct measurement of absorbed dose was later developed by ROSSI and FAILLA.[5] Since these chambers also serve as γ-ray dosimeters, provisions must be made for separating the two types of radiation. This is accomplished by a subtractive method where γ-ray dose is determined with a second dosimeter usually constructed of carbon or beryllium. The neutron efficiency of such chambers is of the order of from 10 to 20 per cent.

* In accordance with Navy requirements, we affirm that the opinions or assertions contained in the paper are those of the writers and not to be construed as official or reflecting the views of the Navy.

† Present address: University of California Medical School, Los Angeles 24, California.

A fast neutron proportional counter, developed by HURST,[6] discriminates against γ-radiation and allows a direct measure of neutron dose. An ethylene-filled proportional counter surrounded by a polyethylene wall provides an accurate measure of dose over a wide range of neutron energies. γ-Background is eliminated by setting the pulse height discriminator sufficiently high to exclude any γ-ray pulses. By means of a pulse integrating circuit, the counting rate of the instrument can be made proportional to neutron ionization.

Neutron dosimetry at the 60 in. cyclotron was carried out using the second approach which consisted of measuring fast neutron flux with sulfur threshold detectors and neutron spectrum with nuclear emulsions. From considerations of scattering cross-section and the chemical composition of tissue, the first-collision neutron dose can then be calculated. A detailed account on the method of determining first-collision dose based on neutron flux and spectrum measurements has been given by ROSSI.[7]

Sulfur wafers of 1 in. diameter were calibrated by making known exposures to 14.1 MeV neutrons at Los Alamos Scientific Laboratory and then counting the detectors in a standard counting geometry. The cross-section for the reaction $S^{32}(n,p)P^{32}$ has been given by KLEMA and HANSON[8] for energies up to 5.8 MeV. Above 4 MeV, a constant cross-section of 300 mbarns was obtained.

Calculations of neutron dose obtained with the 20 MeV deuteron beam were made assuming the sulfur cross-section to remain constant from 4 to 20 MeV.[3] ALLEN *et al.*[9] have recently extended the sulfur cross-section measurements to neutron energies from 1.6 to 15 MeV. From their data, a more accurate evaluation of first-collision neutron dose has been made. As a result, the values given in previous reports should be revised downward by 19 per cent.[1,2,3]

ROSEN[10] has presented a detailed review of the various track plate techniques available for measuring neutron energy spectra. Following the procedures described by ROSEN, Ilford C-2 nuclear track plates 200μ thick were used to measure the three neutron spectra given in this report.

The production of cyclotron neutrons is always accompanied by γ-radiation. γ-Rays are produced by interaction of the uncollimated components of the circulating particle beam with the dees and deflector plate. The production of neutrons at the target also gives rise to γ-radiation. This results primarily from decay of excited levels of the product nuclei to ground level. The γ-rays from the target are proportional to neutron flux and can be measured independent of the normal cyclotron γ-ray background by extending the beam and target outside the cyclotron room.

In order to determine cyclotron γ-ray background with photographic emulsions it was first necessary to evaluate the sensitivity of film to the particular neutron energies in question. For this purpose, a study was made on the response of film to charged particles over a wide range of specific ionization.[11] Protons, deuterons and alpha particles of 340, 190, and 380 MeV respectively were obtained from the 184 in. frequency-modulated cyclotron at the University of California Radiation Laboratory. The specific ionization of the various beams could be varied by means of absorbers to provide a range of ionization extending from 1.6 times the minimum to the maximum value for singly charged particles. From these data, and from direct film exposures to the various neutron spectra investigated, the neutron sensitivity of several emulsions was obtained. The relatively low sensitivity of film to cyclotron neutrons (0.05 to 0.01 as sensitive as γ-rays) greatly facilitated measurement of γ-ray background.

PHYSICAL CHARACTERISTICS OF CYCLOTRON BEAM

Axial and transverse neutron distribution

Given in Fig. 1 are the sulfur neutron flux measurements along the beam axis for both 12 MeV protons and 24 MeV deuterons striking a thick Be target. Both distributions follow the inverse square law which indicates the absence of significant fast neutrons from other parts of the cyclotron.

Because sulfur is more efficiently activated by neutrons from the deuteron beam it is more informative to compare the two outputs directly

in terms of neutron dose rate. These data are presented in Table 1 along with similar information for the earlier 20 MeV deuteron spectrum. The neutron rad/min, given in column 4, is based on the first-collision neutron rad dose

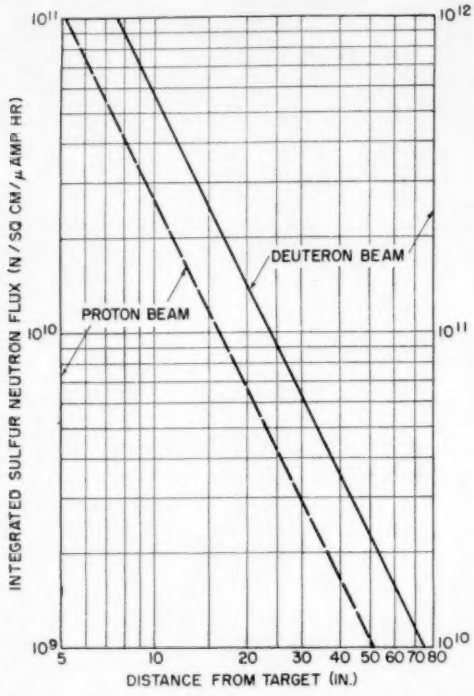


FIG. 1. Axial sulfur neutron flux for 12 MeV protons and 24 MeV deuterons on a thick Be target.

for the given spectrum and on the efficiency of the sulfur detector relative to 14.1 MeV neutrons.

Figs. 2 and 3 show the transverse neutron distributions at various distances from the target for the two neutron spectra. The deuteron neutrons are seen to be highly directional in comparison to the lower energy proton neutrons. For irradiation purposes it is usually important to obtain a reasonably large and uniform field distribution. This consideration favors the proton beam since at any given distance from the target a relatively larger irradiation area will be available. Approximately the same field geometry is obtained at 20 in. for the proton beam as at 50 in. for the deuteron beam. The relative intensities given in Fig. 1 should be

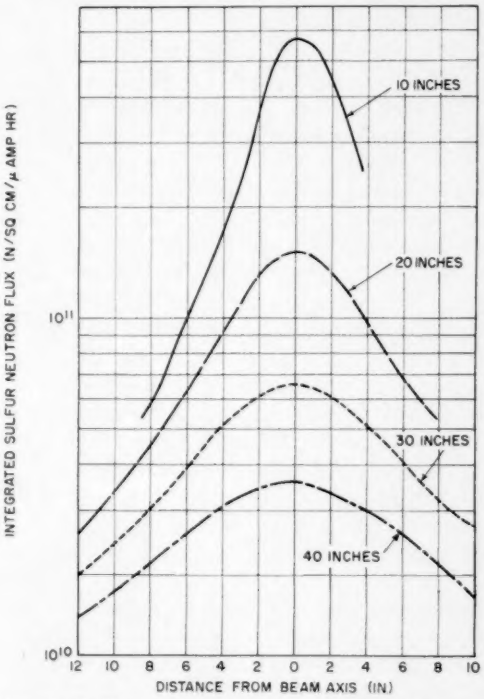


FIG. 2. Sulfur neutron flux transverse to beam axis for 24 MeV deuteron beam at various distances from target.

Table 1. *Neutron flux and dose rate* 30 in. *from the target for deuterons and proton beam on* Be *target*

Beam	Sulfur (n/cm^2 per μA-hr)	Sulfur (n/cm^2 per rad)	Neutron rad/min at 60 μA (rad/μA-hr)
12 MeV Protons	3.0×10^9	9.9×10^7	30
20 MeV Deuterons	3.4×10^{10}	2.4×10^8	140
24 MeV Deuterons	5.5×10^{10}	2.2×10^8	250

compared on this basis, rather than on peak intensities whenever similar field distributions are required.

Neutron spectra

By means of a neutron camera technique and nuclear track plate measurements, the neutron spectra for both 20 and 24 MeV deuterons on Be have been determined. The camera is an evacuated cylinder with a thin polyethylene radiator located on one end. A collimated beam of neutrons is allowed to strike the radiator and pass through the cylinder without exposing the track plates which are positioned in the camera so that protons recoiling from the radiator at angles of approximately 5° will be intercepted. Use of the camera considerably simplified track analysis because only those protons originating at the top surface within specified angular limits would produce valid tracks.

Results of the neutron spectral measurements obtained with the two deuteron beams are given in Fig. 4. For comparative purposes, the spectrum obtained by COHEN and FALK[12] for 15 MeV deuterons on Be is also included.

The neutron spectrum produced by bombarding a thin Be target with 16 MeV protons has been determined by GUGELOT[13] using nuclear emulsions. Based on these data, SHEPPARD and DARDEN[14] have estimated the distribution of neutrons emitted by 22 MeV protons on a thick Be target at the Biological Facility of the Oak Ridge National Laboratory 86 in. cyclotron. These studies showed that neutrons produced by a proton beam are of considerably lower energy than those produced by deuterons and further indicated that, with the proper choice of proton energy, a spectrum similar to fission neutrons could be obtained.

Since a fission neutron spectrum was of direct interest in the investigations being conducted, measurements were carried out at the Berkeley 60 in. cyclotron to determine the neutron spectrum from 12 MeV protons on a thick Be target. Fig. 5 shows the cyclotron spectrum together with: (a) the primary fission neutron spectrum as determined by WATT,[15] (b) the leakage spectrum from the Los Alamos "Godiva" critical assembly as measured by ROSEN,[16] and (c) the neutron spectrum in the Biological Facility of the Argonne CP-5 reactor as measured by the authors. Differences in (b) and (c) from the primary neutron spectrum (a) are brought about by degradation of the primary beam. The "Godiva" assembly consisted of two 6 in. hemispheres of U^{235} which represents the minimum mass necessary to sustain fission. In the Argonne facility, neutrons are produced by thermal neutron induced fission in a 1 in. thick uranium converter plate, and must then pass through a 4 in. bismuth shield before entering the biological port.*

It is seen that neutrons produced by a 12 MeV proton beam closely simulate the primary fission

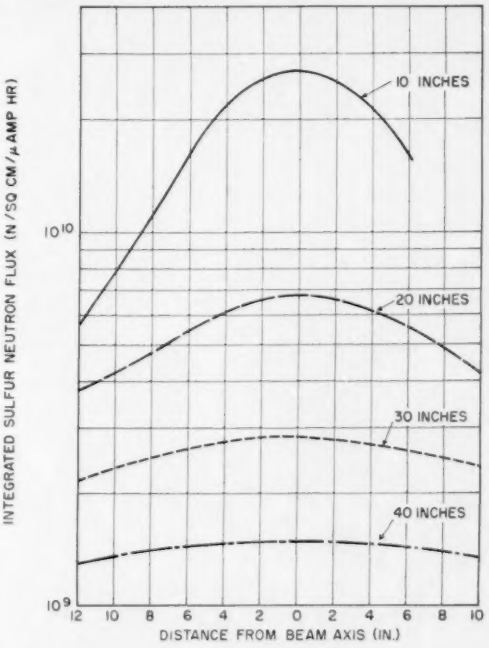


FIG. 3. Sulfur-neutron flux † transverse to beam axis for 12 MeV proton beam at various distances from target.

neutron spectrum. Should it be necessary to approximate any of the degraded spectra, the cyclotron neutrons could be further modified by introducing a suitable moderator.

* Unpublished data by H. VOGEL, Argonne National Laboratory.

† As used in this paper sulfur-neutrons or sulfur-neutron flux refers to neutrons above the sulfur activation threshold.

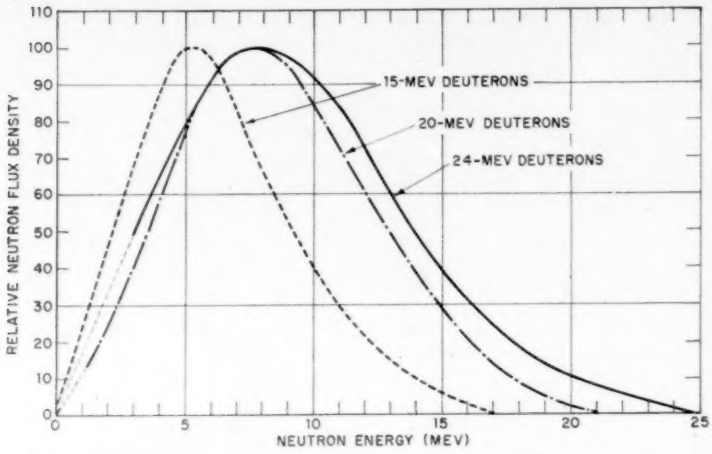


FIG. 4. Forward neutron spectra of 15, 20 and 24 MeV deuterons on Be. The 15 MeV spectrum is taken from the data of COHEN and FALK.[12]

Depth–dose measurements

Given in Fig. 6 are neutron depth–dose curves taken in water for the 24 MeV deuteron beam. The fast neutron distribution was determined with sulfur detectors while the thermal distribution was obtained with gold foils. Fig. 7 is a similar plot for the 12 MeV proton beam. These curves are of interest in determining the amount of thermal neutron flux that may be obtained for any given beam.

The thermal neutron distribution is observed to peak between 1.5 and 2 in. from the surface. Since the thermal neutrons are produced by slowing down of the fast neutrons in hydrogen, this peak is independent of distance from the target. Similar curves would also be obtained in plastic or other hydrogenous material, the degree of moderation being determined by their relative hydrogen densities. The peak thermal flux in Figs. 6 and 7 is proportional to the sulfur flux at the surface and for this reason its value in water at any distance from the target can be predicted from the axial sulfur values of Fig. 1. Thus with the phantom located at a distance of 20 in., the thermal flux would be approximately 1.8×10^{11} n/cm^2 per μA-hr for the deuteron beam as compared to a value of 3.7×10^{10} n/cm^2 per μA-hr for the proton beam.

Marked differences between the sulfur and thermal neutron measurements over the first few inches of Figs. 6 and 7 indicate that the spectrum is being strongly modified in passing through the medium. The constant ratio between the gold and sulfur readings beyond several inches has been interpreted as signifying that an

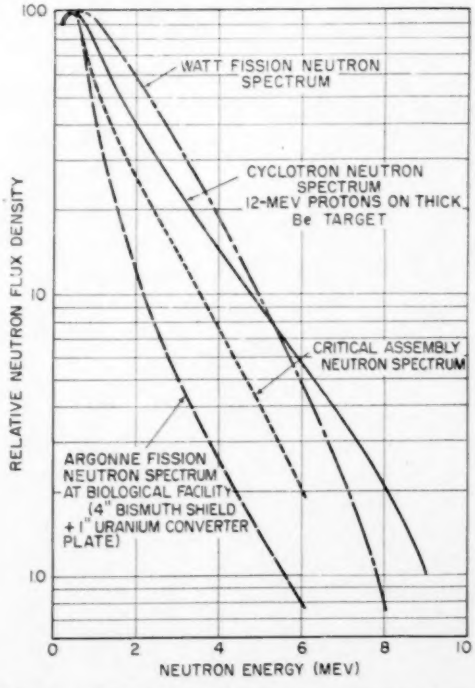

WATT FISSION NEUTRON SPECTRUM
CYCLOTRON NEUTRON SPECTRUM 12-MEV PROTONS ON THICK Be TARGET
CRITICAL ASSEMBLY NEUTRON SPECTRUM
ARGONNE FISSION NEUTRON SPECTRUM AT BIOLOGICAL FACILITY (4" BISMUTH SHIELD + 1" URANIUM CONVERTER PLATE)
RELATIVE NEUTRON FLUX DENSITY
NEUTRON ENERGY (MEV)

FIG. 5. Comparison of neutron spectrum from 12 MeV protons on Be to three neutron fission spectra.

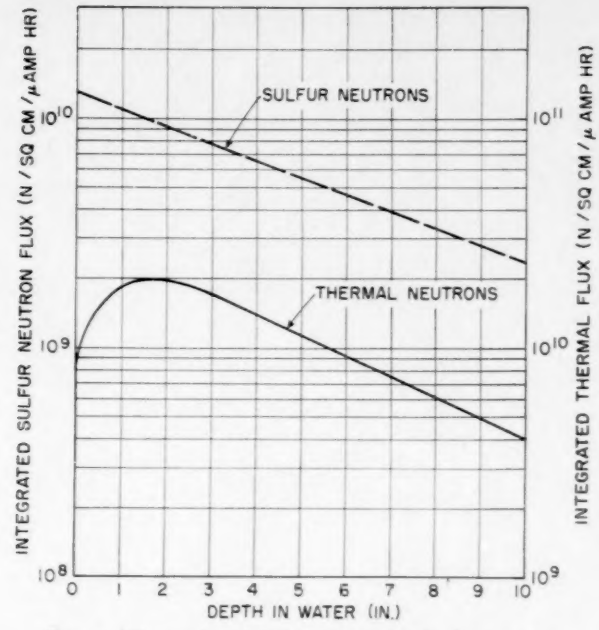

SULFUR NEUTRONS
THERMAL NEUTRONS
INTEGRATED SULFUR NEUTRON FLUX (N / SQ CM / μ AMP HR)
INTEGRATED THERMAL FLUX (N / SQ CM / μ AMP HR)
10^{10}
10^{11}
10^{9}
10^{10}
10^{8}
10^{9}
0 1 2 3 4 5 6 7 8 9 10
DEPTH IN WATER (IN.)

FIG. 6. Thermal and fast neutron depth–dose curve in water 59 in. from target for 24 MeV deuterons beam.

equilibrium condition has been approached where the neutron spectrum is no longer changing significantly with depth.

γ-Ray background

Determination of γ-ray dose with photographic emulsions requires a knowledge of film response to the accompanying neutron spectrum. The neutron-induced component of film blackening from the deuteron beam was determined by a series of transverse film and sulfur measurements made outside the cyclotron room where only those neutrons and γ-rays originating at the target were observed. For this purpose the deuteron beam was piped through the walls of the cyclotron room by means of a collimating tube.

The neutron and γ-ray components of film blackening resulting from 24 MeV deuterons can be readily separated because of differences in their spatial distribution. From Fig. 2 it is seen that the neutrons are highly directional, the beam falling to one-half its peak intensity approximately 16° from the axis. On the other hand, the accompanying γ-rays which result from activated or excited states of the target nuclei can be assumed to be isotropic.

The neutron film sensitivity was obtained by

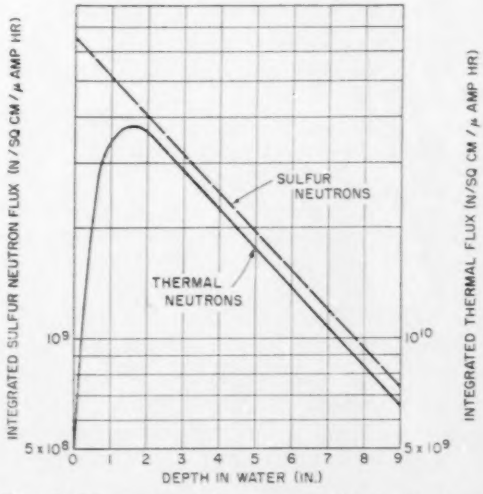

SULFUR NEUTRONS
THERMAL NEUTRONS
INTEGRATED SULFUR NEUTRON FLUX (N / SQ CM / μ AMP HR)
INTEGRATED THERMAL FLUX (N/SQ CM / μ AMP HR)
10^{9}
10^{10}
5×10^{8}
5×10^{9}
0 1 2 3 4 5 6 7 8 9
DEPTH IN WATER (IN.)

FIG. 7. Thermal and fast neutron depth–dose curves in water 20 in. from target for 12 MeV proton beam.

determining what component of neutron film blackening, when subtracted from the total readings, would effect the best fit to an isotropic γ-ray distribution. These measurements indicated that, along the beam axis, the γ-ray ionization was approximately 1 per cent of the neutron rad dose.

Similar transverse measurements were also taken inside the cyclotron room under normal exposure conditions. Table 2 shows the ratio of γ to neutron dose along the beam axis at various distances from the target. These measurements indicate that the γ-rays from within the cyclotron are primarily from the region of the deflector located behind the target and somewhat askew to the beam axis.

Table 2. Ratio of γ-ray dose to neutron dose along beam axis for deuteron and proton beams

Distance from target (in.)	Percent of neutron dose	
	24 MeV deuterons	12 MeV protons
10	2	4
20	4	7
30	5	12
50	7	17

Because the angular distribution of neutrons from the 12 MeV proton beam showed but slight directional properties the technique used with deuterons to determine neutron film sensitivity could not be used with any great confidence. However, additional tests were made by exposing films to the Los Alamos "Godiva" critical assembly whose neutron and γ-ray emission had been previously determined.* These measurements showed that for Eastman Translite film approximately 100 rads of fission neutrons were required to produce the same degree of blackening as 1 r of Co^{60}. This observation is in agreement with the predicted film sensitivity based on studies of film response to charged particles.[13] In column 3 of Table 2 the relative γ-ray background along the beam axis is given as determined by the film technique. Comparison of columns 2 and 3 shows that a considerably greater γ-ray background is associated with the proton beam.

Reproducibility and relative accuracy of physical measurements

For neutrons produced by the original 20 MeV deuteron beam, several comparisons have been made with measurements obtained by other investigators. An animal irradiation study similar to that conducted by CARTER, BOND, and SEYMOUR at the 60 in. cyclotron was performed at Los Alamos using mono-energetic 14 MeV neutrons produced by the (D,T) reaction.[2.17] The same strain of mice was used and identical biological end points (spleen and thymus weight loss) were measured in both cases. The relative neutron exposures differed by 18 per cent when compared on the basis of first-collision tissue dose.

An intercomparison between neutron measurements made with tissue equivalent chambers, proportional counters, threshold detectors, and chemical dosimeters was carried out at the Argonne National Laboratory 60 in. cyclotron.[18] The neutrons were produced by bombarding a thick Be target with 21 MeV deuterons. The results of all measurements showed remarkably good agreement, with the average deviation among the various methods being of the order of 8 per cent.

Changes in the calculated value of neutron dose rate for the 20 MeV deuteron beam, resulting from new sulfur cross-section data, has brought about a revision of previously reported values. These changes are listed in Table 3. Agreement with the Los Alamos biological data is now within 5 per cent. On the other hand, the comparative physical measurements made at Argonne now differ by the order of 20 percent.

The standard exposure procedure at the cyclotron was to rotate a group of samples or biological specimens positioned on a turntable about the beam axis and thereby obtain an identical exposure for the entire group. Sulfur wafers placed on the turntable served to monitor each run. The sulfur detectors were counted no earlier than 24 hr following exposure to allow decay of short-lived activities that were found to be present. Checks on the reproducibility

* Unpublished data by J. A. SAYEG, Los Alamos Scientific Laboratory.

Table 3. *Revised neutron data for* 20 MeV *deuterons on* Be

Type of experiment	Reported values	Revised values
RBE—Mice[2]		
(a) Spleen–thymus weight loss	1.7–1.5	2.0–1.8
(b) Mortality	1.6	1.9
RBE—dogs, mortality[1]	0.81	0.95
Neutron dose USNRDL sulfur detector[20] / Neutron dose ORNL proportional counter	0.94	0.79

RBE values are relative 250 KVP X-rays

of the three neutron beams used over a 3 year period have shown that for any series of runs made on a given day the average flux per μA-hr, as determined by sulfur detectors and a current integrator, normally will not differ by more than a few per cent. However, differences in beam intensities for runs made on different days have varied by as much as 15 per cent. Some of this difficulty arises from the use of different targets; still another cause is the fact that the beam axis may shift slightly from one run to the next. However, such large fluctuations are rather infrequent so that the beam intensity from run to run will normally remain constant to within 5 per cent. This important consideration, together with high neutron intensity, suitable choice of spectra, and low γ-ray background, has made the cyclotron a most useful instrument for controlled neutron exposures.

REFERENCES

1. V. P. Bond, R. E. Carter, J. S. Robertson, P. H. Seymour and H. H. Hechter, *The Effect of Total Body Fast Neutron Irradiation in Dogs*. U.S. Naval Radiological Defense Laboratory Technical Report USNRDL-TR-36 (1955); *Rad. Res.* **4,** 139 (1956).
2. R. E. Carter, V. P. Bond and P. H. Seymour, *The Relative Biological Effectiveness of Fast Neutrons in Mice*. U.S. Naval Radiological Defense Laboratory Technical Report USNRDL-TR-20 (1954); *Rad. Res.* **4,** 413 (1956).
3. E. Tochilin, S. W. Ross, B. W. Shumway, G. D. Kohler and R. Golden, *Cyclotron Neutron and Gamma-ray Dosimetry for Animal Irradiation Studies*. U.S. Naval Radiological Defense Laboratory Technical Report USNRDL-TR-25 (1954); *Rad. Res.* **4,** 158 (1956).
4. P. C. Aebersold and G. S. Anslow, *Phys. Rev.* **69,** 1 (1946).
5. H. H. Rossi and G. Failla, *Medical Physics* (Edited by O. Glasser), Vol. II, pp. 603–607. Year Book Publishers, Chicago (1950).
6. G. S. Hurst, *Brit. J. Radiol.* **27,** 353 (1954).
7. H. H. Rossi, *Radiation Dosimetry* (Edited by G. H. Hine and G. L. Brownell), pp. 667–691. Academic Press, New York (1956).
8. E. D. Klema and A. O. Hanson, *Phys. Rev.* **73,** 106 (1948).
9. L. Allen, Jr., W. A. Biggers, R. J. Prestwood and R. K. Smith, *Phys. Rev.* **107,** 1363 (1957).
10. L. Rosen, *Nucleonics* **11,** No. 7, 32; No. 8, 38 (1953).
11. E. Tochilin, B. W. Shumway and G. D. Kohler, *Rad. Res.* **4,** 467 (1956).
12. B. L. Cohen and C. E. Falk, *Phys. Rev.* **84,** 173 (1951).
13. P. C. Gugelot, *Phys. Rev.* **81,** 51 (1951).
14. C. W. Sheppard and E. B. Darden, *Physical Dosimetry with Fast Neutrons from a Cyclotron Source*. Oak Ridge National Laboratory Report ORNL-1559 (1953).
15. B. E. Watt, *Phys. Rev.* **87,** 1037 (1952).
16. L. Rosen, *Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Geneva*, 1955, Vol. IV, Paper P/582. United Nations, New York (1956).
17. P. S. Harris and L. E. Ellinwood, *Biological Effectiveness of* 14 *MeV Neutrons; Spleen and Thymus Weight Loss in Mice as the Biological Indicator*. Los Alamos Scientific Laboratory Report LA-1629 (1953).
18. R. B. Rhody, H. H. Rossi, G. S. Hurst, E. Tochilin and S. Sigoloff, *Rad. Res.* **5,** 495 (1956).

Health Physics Pergamon Press 1958. Vol. 1. pp. 340–344. Printed in Northern Ireland

A PULSE READING METHOD FOR CONDENSER ION CHAMBERS*

W. C. ROESCH, R. C. McCALL and F. L. RISING
Hanford Laboratories Operation, General Electric Company, Richland, Washington

(*Received* 7 *July* 1958)

Abstract—A new method has been developed for the measurement of low levels of radiation using condenser ionization chambers such as the pencils used in radiation protection work. The method of reading utilizes the magnitude of the current pulse required to recharge the pencils after they have been exposed. Any charging voltage may be used without a change in sensitivity. Using a low charging voltage reduces leakage to a very low level. Full range of pencils charged to 20 V is about 30 mr. Higher charging voltages can be used to increase this range. Sensitivity of a small fraction of an mr is available. Readings of 1 mr $\pm$ 0.2 mr or 10 mr $\pm$ 1 mr are possible at 95 per cent confidence with a single pencil.

The system has been used at HAPO to measure background on the project, on the Columbia River, and in the environs. Typical values are presented.

Other possible applications are accurate personnel monitoring and neutron dose measurement. This system lends itself to automation of records keeping.

I. INTRODUCTION

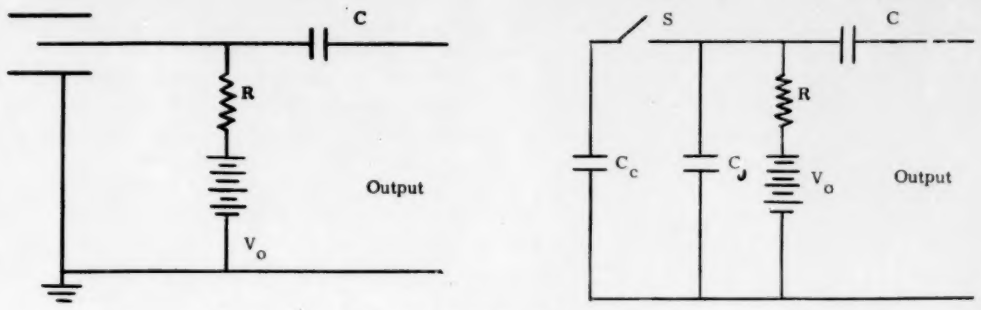
In the usual application of the well known condenser ion chambers,[1] the chamber is first charged to a given voltage V_0, then exposed to radiation, and then the resulting voltage V_1 on the chamber is measured with an electrometer. The dose measured by the chamber is the proportional to $V_0 - V_1$. A typical example of this type of chamber operation is the common pencil dosimeter used in radiation protection work. It will be used as an example throughout this article. This method has several disadvantages. The most important of these is that in general you are measuring a small difference between two large numbers, resulting in large errors in measuring small doses. Also, the reading and recharging operations are separate, making the process rather slow. The process does not lend itself to automation.

Two ideas were conceived and combined in order to correct these difficulties. The first idea was to measure the voltage on the pencil by discharging it through a resistor R. A voltage pulse proportional to V_1 would then be produced across R which could be amplified and measured by any pulse height analyzing system more accurately than with an electrometer. The second idea was to use a precision voltage supply for V_0 and measure the difference $V_0 - V_1$ directly in order to obtain greater accuracy. This could be done with a sufficiently sensitive electrometer. Actually, this idea was combined with the first one and the pencil was recharged to V_0 after exposure through a resistor R. During the recharging process, current flows through R and a voltage pulse is produced proportional to $V_0 - V_1$. This pulse height can again be measured by a variety of methods. In this way most of the disadvantages of the old method can be overcome, and reading and recharging become a single operation.

II. THE PULSE CIRCUIT

In Fig. 1(a), the basic circuit of the pulse reading method is shown. J is the jig into which the pencil fits while being charged; R is the signal resistor; V_0 is the stable voltage supply; and C is the coupling capacitor to the pulse height analyzing equipment. Fig. 1(b) shows the equivalent circuit of this

* Presented at the Health Physics Society Meeting held at Berkeley, California, 9–11 June 1958.



(a) FIG. 1. The basic circuit. (b)

operation. C_c is the capacity of the pencil and C_j is the capacity of the jig and some stray wiring capacity. If a reading circuit is attached to the output, its input capacity should also be included in C_j. Before the pencil is read, C_c is charged to V_1, C_j is charged to V_0 and S is open. At time $t = 0$, S is closed, charge is shared between C_c and C_j and current flows through R until $C_c + C_j$ is charged to V_0 again, producing a voltage pulse at the output.

Before $t = 0$, the charge Q_c on the pencil is $C_c V_1$ and the charge Q_j on C_j is $C_j V_0$. At $t = 0$, C_j and C_c are placed in parallel and the voltage across them V_p is $(Q_c + Q_j)/(C_c + C_j)$ or $(C_j V_0 + C_c V_1)/(C_c + C_j)$. The voltage across R at $t = 0$ is then $V_0 - V_p$ or $\{C_c(V_0 - V_1)\}/\{C_c + C_j\}$. This voltage then decays exponentially with a time constant of $R(C_c + C_j)$. For a typical pencil, C_c is about 5 pF and C_j is somewhat smaller. It is seen that the output pulse is then proportional to $V_0 - V_1$ with the proportionality constant somewhat less than 1.

In our typical 5 pF pencil, a voltage decrease of about $\frac{1}{3}$ V per mr of exposure is obtained. If we assume C_j is also 5 pF, an output voltage pulse of $\frac{1}{6}$ V/mr is produced, or 166 mV/mr. Since 1 mV pulses are easily handled in pulse height analyzing systems, the method is capable of reading doses of less than 0.01 mr. Other considerations turn out to be the limiting factors, as will be seen later.

It is also important to note that for a given pencil and exposure the difference $V_0 - V_1$ is constant regardless of the value of V_0 as long as saturation is obtained over the range $V_0 - V_1$. In many cases then, V_0 can be made quite small, reducing the insulator leakage of the pencils.

III. CIRCUITRY

Since existing commercial equipment will provide for automation of reading and recording, it was desired to obtain a relatively simple circuit for reading the pencils that would give a dial type readout.

Basically, the circuits used can be broken into three separate blocks. First, the input circuit, which is a very high input impedance cathode follower; second, a stable slow-pulse amplifier; and third, a peak reading voltmeter which will measure the amplitude of a single pulse.

The greatest problem encountered was due to the noise of imperfect contact between the pencil and the jig. If the input circuit had sufficiently high input impedance and long time constants, the current was integrated over such a long time that this fast noise was negligible. The original circuits tried had an input impedance of from 1 to 10 MΩ, and a time constant of 1 μsec or less. These gave pulses that had what appeared to be the proper shape, but closer examination with a good pulse oscilloscope showed that the leading edge was very ragged because of imperfect contact when the pencil was inserted in the jig. An attempt was made to increase the input impedance of the conventional cathode follower circuit, but then trouble was encountered because of grid current. This was finally resolved by placing an integrator network in the output of the cathode follower circuit.

This integrated pulse was then fed into a pulse amplifier, and from that into a pulse stretcher and peak reading voltmeter. This circuit was stable, worked well, but had a limited range of input.

A further attempt was made to remedy the low input impedance problem by the use of electrometer tubes. This was successful, but limited the range of the circuit even more severely. The Driven Anode Electrometer circuit[2] was tried, using $1.5 \times 10^{10}\ \Omega$ input and grid resistors. This circuit had sufficient range and still kept the pulse width sufficient that insertion noise was no problem. The pulse width at the output of the input circuit is now about 10 msec, using a pencil of about 5 pF capacity. This circuit (Fig. 2) has been used for input pulse amplitudes of up to 140 V, giving a range of about 370 mr to a Victoreen No. 362 pencil. This can be compared to the 200 mr full range of the Victoreen Model No. 240 Projection String Electrometer which charges the pencil to 150 V. The calibration of this electrometer can be extended to obtain a range of at least 280 mr, however.

The amplifier and peak reading voltmeter presently being used, Fig. 3, is a modification of a circuit[3] originally designed for measuring the intensity of microsecond X-ray pulses. A conventional, heavily filtered and regulated power supply is used to supply the +255, −225, and +300 V required for the circuit. For the pencil charging voltage during development work, a KIN-TEL Absolute d.c. Power Supply was used because of its stability and ability to be reset to exactly the same voltage after long periods of time. For a similar instrument being used for regional monitoring, a stack of mercury cells is used as a stable charging voltage source. Standard cells could be used for a more stable source if necessary.

IV. PROPERTIES OF THE CIRCUIT AND CHAMBERS

The response of the circuit as presently used is not linear, but probably could be made more nearly linear by judicious choice of operating points and feedback. A plot of output vs. dose is given in Fig. 4. The Victoreen Model 362 pencils have a response of 2.75 mr/V decrease in potential on the pencils. Landsverk Model L-65 give about 2 mr/V, and Keleket Model K300 slow neutron pencils have a ratio of 5.4 mrems/V. Victoreen *R* meter chambers have also been used successfully, as have the "*S*" chambers (large volume–very thin wall) formerly used for regional monitoring at Hanford.

Leakage of the Victoreeen Model 362 pencils when charged to 20 V causes a voltage drop equivalent to 1 mr/month, or about 2 μr/hr. It is, of course, higher at higher charging voltages. The leakage was determined by measurements at a fixed dose rate with different air pressures in the chambers and extrapolation to zero pressure.

Minimum measurable dose using the Victoreen pencils is less than $\frac{1}{2}$ mr and is governed by the quality of the pencils used, and the amount of electrical noise in the reader circuit. It has been found necessary to keep the insulators in the pencils tight to prevent them from causing erratic readings. Contact potentials have also caused problems. Since the pencils have aluminum contacts, all of the parts of the pencil jigs were made of aluminum to minimize this problem.

The Victoreen Model 362 pencils have been found to be useful down to a remaining potential of 10 V. With an initial potential of 20 V, this gives a useful range of about 27 mr. Increasing the initial potential to 30 V increases the useful range to about 55 mr. The pencils have been charged to voltages as high as 150 V and read successfully. This gives a useful range of about 380 mr. With these pencils charged initially to 20 V, saturation is obtained for dose rates at least as high as 17 mr/hr. To insure saturation for applications where high dose rates may be encountered, higher initial voltages should be used.

The pencils show very good stability in use. They are quite resistant to shock if the insulator keepers are kept tightly in place, and show little change because of thermal cycling.

One problem that has been encountered is electrical strain on the insulators. This is a characteristic of the pencil and is independent of the method of reading. It has been found necessary to charge a pencil to its initial voltage a number of times over a period of 1–2 days to

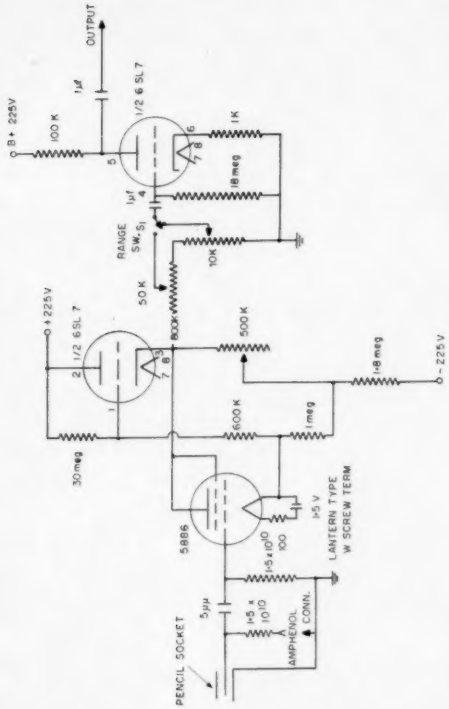


FIG. 2. Output circuit.

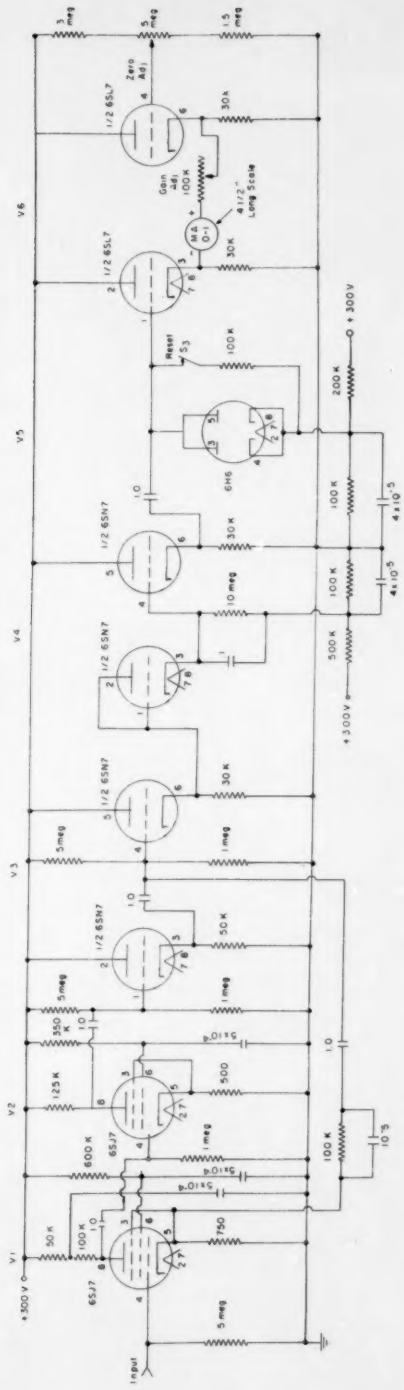


Fig. 3. Amplifier and peak reading voltmeter.

stabilize it if it had previously been charged to some other voltage. Mechanical strain will also cause a change in voltage to appear on the pencil.

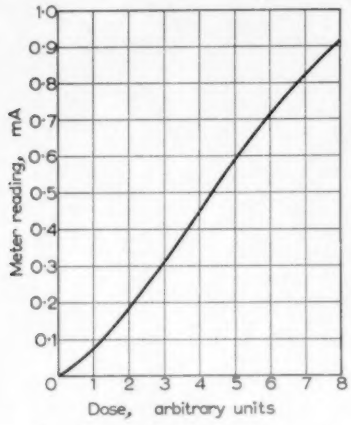


FIG. 4. Calibration curve using Victoreen Model 362 pocket ion chambers.

V. APPLICATION

At Hanford this system has been used to measure dose in the region of 0.2 mr to greater than 18 mr integrated over a period of from 1 day to 2 weeks or more. This range can be increased by changing the charging voltage on the pencils and gain of the instrument. The radiation measured consists of cosmic, natural background, bomb fallout and the component contributed by Hanford in certain areas.

In considering other applications, it might be well to review the special advantages of pencils as detectors. First, the only equipment that actually has to be at the measurement point is the pencil itself. Second, the pencil is a fairly sensitive instrument, allowing the measurement of $\frac{1}{2}$ mr doses with a typical pencil. Third, at the sacrifice of sensitivity the pencil can be made very small or, conversely, at the sacrifice of small size, very sensitive. Fourth, the detector itself, i.e. the pencil, is inexpensive (in large numbers, a few dollars each) allowing the use of large numbers of detectors. Only a few readers are necessary at a cost of several hundred dollars each. Bearing these advantages in mind, many other applications can be imagined.

An obvious application of the pulse reading method is the reading of personnel dosimeter pencils in a large atomic energy plant. Reading, recharging and recording on IBM cards could be accomplished in a single operation. In addition, the sensitivity and accuracy of the pencils for low level measurements could be improved.

Of special interest at this time is the dose delivered by fallout and the degree of protection afforded by different types of buildings. Here the low cost and small size of the pencils would allow them to be placed in large numbers through the environment. Since readings of 1 mr $\pm$ 0.2 mr are possible at 95 per cent confidence level with a single pencil, a 20 per cent increase in background could be measured after about 4 days exposure. Averaging the results of several pencils would increase the accuracy and reliability. β-ray sensitive pencils could be used as well as the usual γ-ray dosimeters.

In biological research it is often very difficult to know the dose received by a particular organ. Since these doses are often quite high, perhaps several roentgens per week or more, special chambers could be made small enough for implanting within the tissues and still retain sufficient sensitivity. In many experiments, the limiting factor on the amount of radioactive material that can be used is the dose received by the gastrointestinal tract. Small pencils could be passed through the animal and the dose read upon recovery.

The pulse reading method is also being applied at Hanford to the measurement of slow neutron dose with boron lined pencils, or of fast neutron dose by the use of suitable moderators surrounding the pencil.

REFERENCES

1. G. J. HINE and G. L. BROWNELL (Editors), *Radiation Dosimetry*. Academic Press, New York (1956).
2. M. H. ARONSON and C. F. KEZER, *One Hundred Electronic Circuits* Vol. I. Pittsburgh Instrument Publishing Company (1957).
3. J. S. ALLEN and D. E. HUDSON, AECD-2052 (1945).

Health Physics Pergamon Press 1958. Vol. 1. pp. 345–351. Printed in Northern Ireland

ELECTRONIC DATA PROCESSING AND RADIATION EXPOSURE RECORDS

I. C. NELSON
Hanford Laboratories Operation, General Electric Company, Richland, Washington

(*Received* 10 *July* 1958)

Abstract—A description is given of an application of electronic data processing methods to the maintenance of Radiation Exposure Records at the Hanford Atomic Products Operation. Described are: magnetic tape records, punched card inputs, computer dose determination, reports, relative costs and record storage. One year's successful operation has proven the worth of the application.

At the Hanford Atomic Products Operation, the Exposure Evaluation and Records Operation maintains radiation exposure records on about 14,000 individuals annually. The individuals included are contractual construction, government and visitor personnel, as well as those regularly employed by the General Electric Company at Hanford.

The principal and by far the most frequent entry to these records is in the form of data from personnel film badge dosimeters. Film badges are processed on a biweekly basis for each of the eight Hanford facilities. On this basis, the number of these dosimeters processed in a year amounts to roughly a quarter of a million.

In the past these data were posted to the individual's record by hand. In addition to posting, the preparation of summary and analysis reports was accomplished by hand methods. The number of individuals recorded and methods of recording and reporting precluded any degree of versatility in exposure analysis.

With Hanford's acquisition of an IBM 702 electronic data processing machine and associated electronic accounting machines, a modernization of the radiation exposure record keeping system was initiated. This paper describes the resulting modernization.

THE RECORD

Each person entering a zone of possible radiation exposure is required to wear a film badge, and for each such person a master record of exposure is formed and maintained. Fig. 1 illustrates the contents of an individual's master record. As shown, the master record contains personal background information as well as exposure totals.

The radiation or payroll number is the code by which all entries to the record are made. The facility clearance codes show for which facility the individual is cleared for entrance. In the case of multiple clearances, the code is generally for that facility in which the individual most frequently works. The Bioassay sampling frequency codes show for which months of the year a bioassay sample is required.

The inclusion of the personal background information permits the use of the record in developing useful exposure statistics and gives the system flexibility in producing exposure related items, such as labels for bioassay sample containers.

The master record reflects by month the biweekly film badge dosimeter results for both β and γ. The principal reasons for retaining the individual month totals are (a) to provide the means of producing the 12-months-to-date total and (b) to simplify exposure audit procedures. The 12-months-to-date total is used by exposure control personnel principally to indicate a current annual exposure and to predict early in any calendar year an individual's probable exposure for that year.

RADIATION OR PAYROLL NUMBER	ORGANIZATIONAL SUFFIX	JOB CLASS OR CRAFT	SOCIAL SECURITY NUMBER	NAME	HOME ADDRESS

BIRTH DATE	SERVICE DATE	TERMINATION DATA	SEX	FACILITY CLEARANCE CODES	BIOASSAY SAMPLING FREQUENCY CODES

JANUARY 1958 EXP.		FEBRUARY 1958 EXP.		MARCH 1958 EXP.		APRIL 1957 EXP.		MAY 1957 EXP.		JUNE 1957 EXP.	
BETA	GAMMA	BETA	GAMMA	BETA	GAMMA	BETA	GAMMA	BETA	GAMMA	BETA	GAMMA

JULY 1957 EXP.		AUGUST 1957 EXP.		SEPTEMBER 1957 EXP.		OCTOBER 1957 EXP.		NOVEMBER 1957 EXP.		DECEMBER 1957 EXP.	
BETA	GAMMA	BETA	GAMMA	BETA	GAMMA	BETA	GAMMA	BETA	GAMMA	BETA	GAMMA

CALENDAR YEAR EXPOSURE			TWELVE MONTHS EXP.		ACCUMULATED EXPOSURE		
BETA	GAMMA	X-RAY	BETA	GAMMA	BETA	GAMMA	X-RAY

OFF-SITE EXPOSURE		LAST EXP. ENTRY		DATE OF LAST EXP. ENTRY
BETA	GAMMA	BETA	GAMMA	

FIG. 1. Individual exposure record format.

As the heading implies, the calendar year totals reflect the exposures accumulated in the current year. At year-end the calendar year total and 12-month-total are equal. The X-ray exposure shown is composed of exposures to a small group of photon energies whose mean is 17 keV. The separate calculation of this dose is useful in controlling γ-exposures. This is not the exposure received from medical X-ray examination or treatment. Accumulated exposure represents the total known industrial exposure and includes exposures received at locations other than the Hanford Works. Off-site exposure is that received from the other locations. The last dose entry and its date are useful in determining the activity of an individual in radiation zones.

The record described is included on a magnetic tape record 266 characters long. This information occupies 2.08 in., including a 0.75 in. record gap, of magnetic tape. The magnetic tape reels presently in use allow about 14,000 such records per reel. For storage purposes 14,000 such records may be stored in 192 in^2.

CARD INPUTS

Parameter cards

The lead card of every processing is the "main parameter" card which has the function of specifying certain machine operation parameters incident to the processing to follow. It gives the identification of the cards being processed, that is, which facility's cards are being processed, and in addition specifies the ending date of the period covered. It further specifies which facility's prepunched badge cards are required for the next processing and the ending date of the period covered by the next processing. The card also contains coding for certain special processing which may be required on a frequency less than that of routine processings. It also contains two dates for file control, such that a machine check may be made to insure that the correct input master file is being used. Several master files or tapes of previous processings are retained and used cyclically to insure recovery should a master file tape be inadvertently destroyed.

The "batch parameter" cards provide the basis for exposure calculation from film density. With each batch or load of film being developed, a set of films exposed to known amounts of radiation are processed. This calibrated film includes several non-irradiated films such that the environmental effect on the film may be accounted for in determining the density of the badged film. The densities of the calibration film are recorded on the batch parameter cards as is the date, batch identification, film lot, and type of calibration—β, γ or 17 keV X-ray.

The equations relating film density or darkening to radiation exposure are then determined on the IBM 702 by the method of least squares. Once the coefficients of these equations and their identification are stored in memory, the machine is ready to accept individual records of density for conversion to exposure and subsequent recording.

Film badge card

The principal function of the film badge card is to enter the individual's film density readings into the system. In addition to the film densities, the card contains the individual's radiation number, calibration or batch identification, a two digit remarks code to indicate special processing or film peculiarities, the individual's name, company organizational code, the ending date of the badge period covered, the facility with which the film is associated and the facility entrance clearance code. The badge cards, exclusive of density connected information, are prepunched for all those known by their facility entrance clearance code to have film to be processed. Generally speaking, these cards are prepunched during the day's processing preceding the day of badge servicing. A few badge cards are key-punch prepared after badge servicing as the identity of those who have had special entry to a facility becomes known.

Alteration cards

Three alteration cards allow for changes to be made to the master records of any of the information not related to exposure, such as address, bioassay sample frequency, etc. One such alteration card provides, in addition to changing certain indicative information, for the entry of an individual's termination date, and the printing of this record without altering the record. Master records of those terminated are retained in the master file for the remainder of the calendar year in which they are terminated. Since each year's master file is retained indefinitely, the records of those terminated are also retained indefinitely.

Exposure adjustment card

The exposure adjustment card permits the adjustment of individual period exposures or exposure totals. The card also provides for the entry of the adjustment date, facility concerned, and a reason for the adjustment. Only the adjusted totals appear in the master record, but the adjustment and reason therefore are entered on several reports.

Miscellaneous input cards

A "removal" card is available which has the ability to completely remove a record from the file. Such removal automatically provides a printout of the record just prior to removal. An "inventory" card allows the printout of an individual's record without altering the record. This card provides for individual accessibility of information on a basis approaching hand filed records.

PROCESSING

To illustrate the flow of information in this system, Fig. 2 presents a general flow chart of the system. The cycle is started with the prepunched badge cards of a previous processing. These cards are merged with those temporary badge cards punched from temporary badge assignments as established by badge servicing. The merged cards enter the Photometry Operation where the badge film is being read on a densitometer. Since a large fraction of film yields a zero net density, those cards representing zero film are separated from those representing reading film. The Photometry Operation also provides the densities of the calibration film. Save for the prepunched information on the badge cards, the key-punching of all input cards is done by personnel of the Exposure Evaluation and Records Operation. The film having zero density is treated specially by automatically key-punching a single character into a certain card column of the badge card which causes the IBM 702 machine to consider such density as zero. Since automatic key-punching requires only the occasional attention of the operator, the process of preparing zero density badge cards is quite efficient. It was found that two key-punch machines, in this case the IBM 026 printing punch and one IBM 056 verifier, were sufficient to handle the business.

The cards are then transported to the electronic data processing machine where about 40 min of machine time completes the processing.

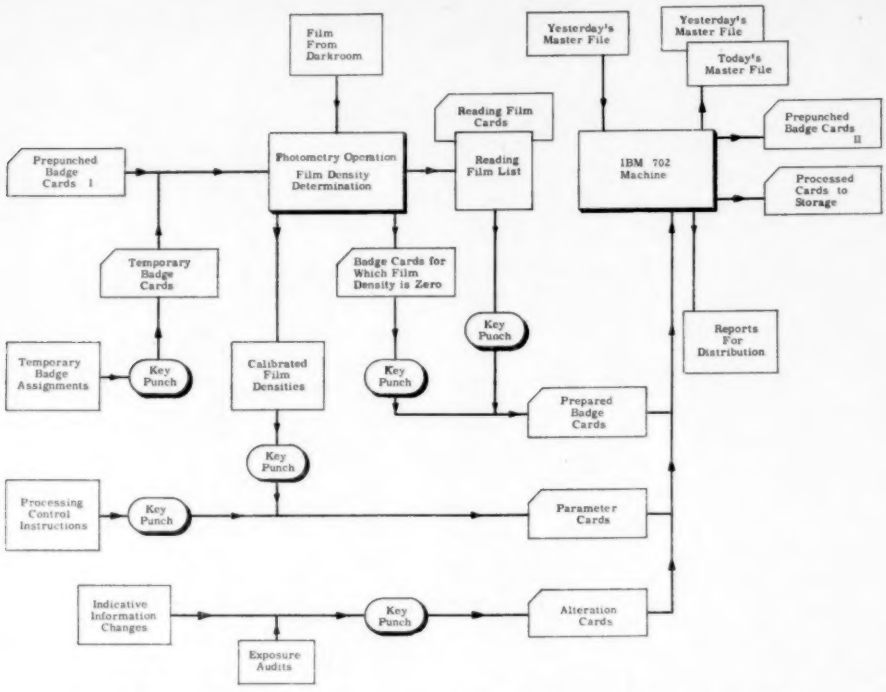


FIG. 2. Generalized flow chart; processing of film badge dosimeter data.

The output of the processing yields the reports, a new master file, the badge cards for storage, and a new set of prepunched badge cards, thus completing the cycle.

REPORTS

Sample pages from several routine reports are shown in Tables 1–4. It should be noted that while the data processing machines compute and report exposure to the nearest milli unit of radiation, this is only for machine convenience and is not a true reflection of the dosimeter sensitivity. At Hanford an individual's exposure information is considered private —much like medical information; hence, the obliteration of names on the report samples.

There are two reports which are published primarily for routine exposure control. The first of these is the Suffix Detail Report and is shown in Table 1. This report provides a listing of all persons and their exposure summaries who are assigned to certain organizations whose work has an increased probability of radiation exposure. It is issued to those exercising exposure control at such time as their individual facility is being processed.

The second such report is the Field Exposure Report and is shown in Table 2. This report lists those persons whose calendar year β-plus γ-exposure meets certain progressive exposure criteria. Thus, a person is reported when his calendar year total exposure reaches or exceeds 300 mrads; again at 600 mrads and so on in 300 mrads increments. In addition, any person having irregularities in film badging or an exposure adjustment will be indicated, regardless of exposure amount. The report is issued at each processing and is independent of facility being processed.

The Suffix Detail Report then provides knowledge of exposure for groups known to require exposure control and the Field Exposure Report provides knowledge of all other significant exposures. In addition, the Suffix Detail Report

Table 1

200 W Area — Detail Suffix Exposure Report — Period Ending 17 February 1958

Suffix	Pay #	Last activity β	Last activity γ	Last activity X-ray	12 months total β	12 months total γ	Cal. yr. to date β	Cal. yr. to date γ	Cal. yr. to date X-ray	All time total β	All time total γ
6950	01766									5090	1640
6950	03838		20			440		136		3630	2188
6950	05666		40	45	50	429		266	110	5435	2489
6950	15289		30	12		270		88	15	820	848
6950	15465		40	14		1091		57	16	4020	3573
6950	18708		38			1354		200	7	3445	4048

Table 2

Operations — Field Exposure Report — 100 K Area — Period Ending 16 February 1968

Pay #	Suffix	Batch No.	Last activity β	Last activity γ	Last activity X-ray	Note code	12 months total β	12 months total γ	Cal. yr. to date β	Cal. yr. to date γ	Cal. yr. to date X-ray	All time total β	All time total γ
17005	5950	0052		111			83	2035			403	3848	8832
18816	5970	0052		121			57	1728			625	1125	6062
19158	5960	0035		104		19	176	1659		14	456	1620	4333
19771	5960	0042		69		23	451	2214		113	696	1186	2694
20267	5960	0052	12	212			54	809		46	631	1144	3914
20350	5970					21	24	1383		6	153	279	1548

Table 3

100 K Area — Film Exposure Statistics — Period Ending 16 February 1958

Suffix	No. of film proc.	No. of zero film	No. of read film	Total dose to file β	Total dose to file γ	Total dose to file sum	Ave. all film β	Ave. all film γ	Ave. all film sum	Ave. reading film β	Ave. reading film γ	Ave. reading film sum
592	47	17	30	5	2229	2234	0.1	47.4	47.5	0.1	74.3	74.4
593	47	10	37	6	2308	2314	0.1	49.1	49.2	0.1	62.3	62.4
594	85	85					0.0	0.0	0.0	0.0	0.0	0.0
595	23	6	17	17	2380	2397	0.7	103.4	104.1	1.0	140.0	141.0
596	156	105	51	71	4420	4491	0.4	28.3	28.7	1.3	86.6	87.9
597	75	38	37	17	1874	1891	0.2	24.9	24.1	0.4	50.6	51.0

Table 4

100 K Area — Personnel Exposure Statistics — Period Ending 16 February 1958

Suffix	0–50 β	0–50 γ	0–50 sum	51–100 β	51–100 γ	51–100 sum	101–200 β	101–200 γ	101–200 sum	201–300 β	201–300 γ	201–300 sum	301–400 β γ sum	401–500 β γ sum	501–1000 β γ sum
537	27	24	24	2	2					1	1				
555	7	2	2	1	1		4	4							
595	23	7	7	2	2		12	12		2	2				
596	156	121	121	19	18		15	16		1	1				
597	75	59	59	14	14		2	2							
825	49	49	49												

provides the basis from which the individuals within a group may be rotated on work assignments such that the individual's radiation exposure may be minimized.

The function of the Personnel Exposure Report, whose format is the same as that of the Field Exposure Report, is to provide a listing of all persons having a non-zero exposure result for the badge period. The batch number is a ready reference of the exposure to the film developing process and its attendant calibration film. The "last activity" provides the latest non-zero result of an individual badge during the reporting period. The note code symbolizes any peculiarity to the exposure entry. The principal use of the Personnel Exposure Report is internal to the Exposure Evaluation and Records Operation for audit and process control purposes.

The remaining reports shown are somewhat of a statistical nature and have principal use within the Exposure Evaluation and Records Operation. The Film Exposure Statistics Report is shown in Table 3 and is self-explanatory. The Personnel Exposure Statistics Report shown in Table 4 gives the number of exposures in various categories for the various organizational suffixes.

Several minor reports of a processing control nature, such as the equations relating film density to exposure, a tabulation of film badge irregularities, etc., are also formed.

With exposure information on tape, the scope of quarterly, annual and special reports is limited only by imagination and funds. This versatility in reporting ability is perhaps one of the most important advantages of this type of system.

One disadvantage of the system is the fact that the information is not as accessible as in a hand filed system. In order to alleviate this condition, the inventory ability previously described was developed. Another aid in handling the records is the Annual Exposure Book. This book contains in condensed form the entire file of master records. This precludes the necessity of processing a past year's file when performing an audit.

SERVICES

Certain other services are provided by having such master records on tape. One such service produces the sampling labels and analytical sheets for the Hanford Bioassay program. These are produced monthly according to the individual's sampling schedule. Another service is the printing of γ-pencil Kardex cards for recording of pencil dosimeter readings.

PROCESSING COSTS

A discussion of costs of this system is perhaps best accomplished in terms of people and machine requirements, since actual costs in any large organization become somewhat complex.

Exposure records operation

When the system was initiated four persons were required in key-punching some of the

historical information for the master records. Once this was accomplished and the system running smoothly, only two persons were needed to effect all phases of input preparation, auditing, and the like. The time of key-punching and verifying the various cards amounts to roughly 2 man-hr per facility processing. At HAPO, four facilities are processed per week. The operators' efficiency increased to the point where data key-punching tasks not directly connected with Exposure Records could be assigned without interfering with their primary duties. It should be pointed out that the operators effect significant control on the system, in addition to simply key-punching data.

Data processing operation

Here again the costs can be only broadly indicated. As mentioned, about 40 min of 702 machine time are required to process a facility's input. Since most installations are utilizing their data processing machines to the maximum, the 40 min is a small fraction of the Data Processing Group's costs. Of course, this system would be out of the question if the total cost of data processing machines had to be charged off to the volume of business described. For installations not having an IBM 702 or similar data processing machines, the use of electronic accounting machines may provide the best path to automation.

Materials, storage

One of the principal benefits of data processing of records lies in smaller storage requirements. As mentioned previously, tape is a very compact means of record storage. Since the cost of tape is quite small, retention of two copies of each year's master file in different locations is a worthwhile precaution.

At this site the input cards enjoy only temporary retention. Since the cards are only a means of converting data into machine language, once they are processed they are of little worth. The cards are retained until the year-end audits have been satisfactorily made and they are then destroyed. At least one copy of all currently published reports is retained permanently.

CURRENT DEVELOPMENTS

At this writing a program modification is being developed that will increase the usefulness of the system. One advance is in the concept of exposure reporting. Instead of the present beta and gamma representation, the new system will include neutrons and be categorized as Derma dose which includes β, γ, neutrons and the 17 keV X-ray exposures; Penetrating dose including γ, neutrons and a portion of the 17 keV X-ray exposures taken to be penetrating; γ including regular γ and a portion of the 17 keV X-ray. This categorization is considerably closer to the nationally recommended treatment of exposures than that presently in use. Another significant modification is in the categorization of the Suffix Detail Report. In addition to breakdown by suffix, this report will now be published further broken down by craft. This will provide exposure control with the optimum in personnel vs. actual work categorization.

CONCLUSION

This system has been in operation since 1 January 1957. The consensus is that the system is satisfying the original objectives of economical exposure record maintenance, increased rate of dissemination of exposure information and an increased scope of reporting information.

Health Physics Pergamon Press 1958. Vol. 1. pp. 352–356. Printed in Northern Ireland

NOTES

Probability of Human Contact and Inhalation of Particles*

(*Received* 28 *August* 1958)

INTRODUCTION

A SITUATION of potential radiological concern is that of radioactive particulate contamination in outdoor work areas of plants employing or processing nuclear materials. Hazards associated with contamination on surfaces are difficult to assess and limited objective studies have been made to establish safe contamination levels.[1–5] A valid assessment of the degree of hazard can be made only when the frequency of deposition and particle radioactivity are known, the physical nature and size of the particles are characterized, and the isotopes present identified. In addition, one must take into account the probability of a particle contacting an individual working in the contaminated area. The known presence of a single radioactive particle may be of no concern if the probability of the particle contacting a person can be shown to be extremely low. The risk which may be justified in various situations may be different, however, knowledge of the probability of contact and exposure will assist materially in establishing limits so that an acceptable risk will not exceeded.

Objective

The work to be reported was undertaken to establish experimentally the numerical probability that individuals would contact or breathe particles located in typical work areas. All parameters affecting particle contact probabilities were not measured, since order of magnitude values were desired.

SUMMARY

Experiments were performed using fluorescent zinc sulfide as a simulant for radioactive particulates to determine order of magnitude probabilities of contact with, or breathing by, workers doing typical operations. Attention was given only to outdoor work, or work which might bring individuals in contact with particles initially deposited on outdoor surfaces. Probabilities ranged from 1×10^{-2} to 1×10^{-4} particles per unit area per particle per unit ground area for transfer from ground to clothing per hour. Probability of breathing a particle while performing typical operations involving vehicles showed probabilities ranging from 6×10^{-7}/hr for driving a vehicle to 6×10^{-4} for sweeping a bus when a single particle was present. Other operations and probabilities are discussed.

EXPERIMENTAL

The estimate of particle contact or breathing probability was determined using fluorescent zinc sulfide placed at points of interest in a manner to simulate the deposition of radioactive particles. The ZnS used had a mean particle diameter of 2 μ. Through standard techniques the material was found to have 1.4×10^{10} particles/g discernible at 100× magnification under ultraviolet illumination. The particle density of 4.1 may be somewhat higher than for some particles or carrier material of interest, but it is believed to be near enough to the average density to be a useful simulant. The particle size distribution of the ZnS is unlikely to be identical to that of radioactive particulates of interest in a given incident. Two considerations suggest that the use of these small particles is justified. Being small, they will become airborne more readily, thereby giving a conservative value for the actual probability of contact expected for somewhat larger, perhaps more typical particles. They will also become attached to inert particles of larger size which could also happen in the case of small radioactive particles. Subsequent movement of the trace particle will be influenced by the carrier or host particle. Although the ZnS chosen for these experiments was not an "identical twin" for actual radioactive particles, the simulation is believed to be adequate for the order of magnitude estimates desired.

Two essentially different kinds of experiments were conducted. These were:

(1) Determination of the particle inhalation probability during operations involving vehicles.

(2) Determination of contact probabilities for operations in open field areas such as walking, digging, etc.

* This work performed under Contract No. W-31-109-Eng.-52 between the Atomic Energy Commission and General Electric Company, Richland, Washington.

Particles picked up by vehicles traveling through contaminated zones may constitute a potential hazard to operators and mechanics. Experiments were conducted to evaluate the probability of particles being released in such a manner that they would be airborne and breathed. Four situations were simulated:

(a) Particles deposited on a bus, then the bus swept in the conventional way; release to the air at head level determined.

(b) Particles deposited below the brake pedal. Their release per hour to the operator determined.

(c) Particles deposited below the foot pedals, then the car swept out by hand using a whisk broom. Particles which would have been inhaled determined.

(d) Particles deposited on a tire, then the tire removed and replaced. Probability of the mechanic inhaling a particle determined.

A known weight of ZnS powder diluted with fine talcum was "seeded" into dirt normally present in vehicles. The operation of interest was then performed in or with the vehicle. During the operation a 0.5 ft^3/min air sample was drawn through a 1 in. millipore filter held near the operator's head. The vacuum pump used was a Trico vacuum booster pump which could be operated from a 6 V battery. American Optical Company R2000 respirators with H-70 paper disks were worn by the operator or mechanic. The number of particles on the filter or mask was determined by counting under a 100 × microscope using an ultraviolet light source for illumination.

The test made to measure the probability of breathing a particle initially present on a tire during dismounting and mounting the tire required a somewhat different "seeding" technique. In this experiment the vehicle was driven repeatedly over zinc sulfide–talcum mixture distributed on a driveway covered with paper.

The vehicle was then driven forward into the garage. Using a specially mounted microscope and ultraviolet illumination the number of particles per unit area of tire tread was measured *in situ* repeatedly and an average taken. The tread area was determined from an imprint made of the tread and planimetering this area. After having determined the number of particles present, the tire changing operation was performed with the mechanic wearing a mask and with the air sampler held adjacent to the mechanic's head. Results of these experiments are tabulated in Table 1.

The second group of experiments was performed to estimate the probability of an individual contacting a particle while working in open terrain contaminated with a known number of particles per unit area.

An area 100 ft × 100 ft was roped off in windblown sandy soil sparsely covered with typical desert growth. A weighed amount of ZnS was thoroughly mixed with sand then distributed uniformly throughout the plot. A hand crank seeder was used to distribute the particles to a density of 4.2×10^6 ZnS particles/ft^2. Workers performed various tasks in this plot of ground. Each wore a strip of double-backed sticky transparent tape on the back of his hands, on the shoulders, overall cuffs, knees and across the nose-piece of his respirator. After the exposure period the tapes were carefully removed and the number of particles determined microscopically. Tasks performed were as follows:

(1) Four individuals wearing the tapes were directed to walk through the plot simulating surveying for contamination. Seven man-hr of exposure in this operation were reached.

(2) A laborer (with tapes similarly placed) dug a shallow trench across the seeded plot. These tapes were afterward collected for examination.

(3) A truck loaded with boxes was driven into the plot, unloaded and loaded twice, by two laborers wearing the pickup tapes and respirators.

In the 5 days of tests, winds were light or negligible the first three. Winds to 10–15 miles/hr were experienced in the ensuing days.

The tapes were examined microscopically to determine the particle transfer probability to the various portions of clothing. Particles collected on filters were too few to be analyzed with significance. Results are shown in Table 2.

RESULTS

The results of the experiments performed are tabulated in Tables 1 and 2. The values tabled are the fraction of particles found on the collector of those initially deposited in the area of interest. For the purpose of this work this number is also equal to the probability of contacting or inhaling a single particle initially present during the interval considered.

After changing the tire the vehicle was driven for 15 miles and the tire again examined to determine the number of particles remaining. The smooth tire retained only 0.0004 of the original number and the new tire retained 0.008 of the original number, indicating that the particles were lightly held.

DISCUSSION

The results tabled give probabilities for contacting small particles initially deposited in outdoor areas. These probabilities could be changed quite markedly by many variables not evaluated in the experiments performed. For example, rain, snow, heavy winds

Table 1. Indicated probability of inhaling a single particle

Operation		Probability	
Driving for 1 hr with a particle on the front floor:			
(a) With windows closed.		6×10^{-7}	
(b) With high air flow through vehicle	(1)	2×10^{-6}	
	(2)	2×10^{-5}	
	(3)	2×10^{-6}	
Note: Run (2) made immediately after placing powder in sedan. Runs (1) and (3) were made the day after placing of particles in the sedan.			
Sweeping floor boards for 1 hr			
(a) Respirator filter count		6.4×10^{-4}	
(b) Using H-70 filter at 0.5 ft³/min		3.3×10^{-4}	
(c) Using MP filter at 0.5 ft³/min		6.2×10^{-4}	
Sweeping a bus for 1 hr			
Average of five experiments (error is standard deviation)		$(4.8 \pm 0.05) \times 10^{-4}$	
		Smooth tire	*New tire*
Tire replacement:			
(a) Respirator filter		2.8×10^{-5}	2×10^{-5}
(b) Using H-70 filter at 0.5 ft³/min		2.3×10^{-5}	3×10^{-6}
(c) Using MP filter at 0.5 ft³/min		—	6×10^{-6}

Table 2. Contact probabilities—particles contacted per hour per unit body area for each particle per unit area on the ground

Operation	Cuff	Knee	Hand	Shoulder	Nose
Walking*	0.014 ± 0.002	0.0023 ± 0.002	0.003 ± 0.001	0.0006 ± 0.0001	0.0017 ± 0.0005
Digging†	0.005 (18%)	0.0007 [0.0011]	0.0016 [0.0022]	0.00054 [0.001]	—
Loading‡ unloading a truck	0.0046 (18%)	0.0002 [0.003]	0.004 [0.008]	0.001 (26%)	0.0019 (22%)

* The error quoted is the standard deviation of the mean of seven 1 hr exposures.

† The 1 hr exposure did not permit a mean and standard deviation to be determined. The error in parentheses ( ) is that calculated from $100/\sqrt{N}$, where N is the number of particles counted. The bracketed value [ ] is the average of a Poisson distribution for which the number of particles counted would be obtained only 10 per cent of the time. This method of expressing the reliability was used when fewer than ten particles were counted per sample. With the limited data, the bracketed value would represent a higher probability than that calculated from the actual number of particles found.

‡ The higher value of two 30 min exposures was used to determine the tabled value. The values in parentheses and brackets have the same meaning as under.†

would individually affect the probability of particles becoming airborne. Additional study should be made of the release, inhalation and impaction probabilities for particles of various sizes and shapes. Probabilities tabled, combined with other observations, can be of assistance in establishing the need for control measures in contamination incidents. The seriousness of consequences arising from contacting or breathing a single particle must first be estimated. This evaluation must take into account all the factors known concerning the chemical, physical and radiochemical properties of the isotopes. Account must also be taken of the credible exposure duration. For exterior body surfaces most potentially exposed the duration of exposure will generally be short because of removal by washing, bathing, attrition and other physical actions. Although these evaluations may be qualitative and somewhat subjective in the absence of specific data, they may lead one to a reasonable estimate of potential bodily insult from a particle contact under the conditions anticipated. This information coupled with the probability of contacting a particle will provide a more realistic, objective assessment of the risk. The conclusion may be reached with good justification that certain levels of particulate contamination may not be of concern. For example, in an area in which surveys show one particle per 10 ft^2, a man may walk on the average of 2000 hr before a particle will be deposited in the face area (assumed $\frac{1}{4}$ ft^2). Even with limited knowledge as to the specific damage which may result from a particle contact of the kind anticipated, there is justification to permit entry and work in such an area. High probability of contact on the other hand is an indication that entrance to such areas should be controlled.

CONCLUSION

The data presented can be an aid in establishing contamination level of concern in certain instances. The probability of contact, however, is only one of several factors which must enter into the estimate of acceptable risk which can be taken.

L. SCHWENDIMAN

General Electric Company
Richland, Washington

REFERENCES

1. J. C. BAILEY and R. C. ROHR, *Airborne Contamination Resulting From Transferable Contamination On Surfaces*, K-1088 (1953).
2. J. L. DICK, W. R. HURDLOW and ELLEN M. HIPELI, *Safe Levels Of Contamination From Fission Products*, AFSWC-TN-56-2.
3. H. J. DUNSTER, *The Derivation Of Maximum Permissible Levels Of Contamination Of Surfaces By Radioactive Materials*, AERE-HP-R-1495 (1954).
4. L. C. EMERSON, T. J. BURNETT, J. C. BAILEY, S. R. BERNARD and D. M. DAVIS, Unpublished studies. Oak Ridge National Laboratory.
5. M. EISENBUD, H. BLATZ and E. V. BARRY, *Nucleonics* **12,** No. 8, 12 (1954).

A Projection Microscope for Neutron Dosimetry*

(*Received* 30 *June* 1958)

ONE of the more time-consuming and tedious techniques used in a Health Physics Personnel Monitoring program is the interpretation of nuclear track film for determining neutron exposures. Toward the goal of faster and more accurate results an attempt was made to design and use a projection-type microscope.

The requirements specified were

(1) Speed of reading.
(2) Room lighting above the threshold of dark adaptation.
(3) Ease of operation.
(4) Use in a small area.
(5) Accuracy.
(6) Adaptability.
(7) Low cost.

Commercial units in general failed to meet specifications (2), (5) and (6). The magnification required (not less than $\times 10$ eye and $\times 44$ dry objective = total 440) reduced the transmission to the point that complete darkness was necessary. The general optics prevented definition of shorter tracks. The projection system could not be used as a normal microscope.

The main problem in meeting the requirements was obtaining a good light source. The available sources fell into three groups: First, the normal incandescent filament lamps; second, the arc sources; and third, the electronic sources.

Elimination of the third group was indicated by high cost, excessive floor space, and considerable electronic maintenance.

The second group, the arc sources, was eliminated since the odor and noise was objectionable and rods had to be changed every few hours. The heat generated could not be tolerated in a small area.

The filament sources in general were too small, the maximum of most projectors being 100 W. However,

* Work was performed under the auspices of the U.S. Atomic Energy Commission.

one unit was usable at both 100 W or 300 W. It was this unit that was finally used.

Using the parallel light from the 300 W bulb we had an almost workable system. Two more steps were then taken. First, a metallurgic-type objective was used. The use of this lens eliminated the need of a cover slip, retaining our definition but eliminating two interfaces. The final step was the use of a $\times 10$ hyperplane eyepiece again increasing the transmission. The use of a projecting prism cost a small fraction of the light but was necessary for adaptability. Fig. 1 shows the component parts, the operator, and the projected area visible on the screen.

The system finally used meets all the initial requirements as follows:

(1) Untrained "film scanners" can work for longer periods with more speed on the projected image than on the binocular microscope.

(2) The 1–2 min to "organize" the film, paper, and pencil provides all the time necessary for dark adaptation.

(3) The lack of cover slips and binocular head reduced the cleaning maintenance to the point that adjustment of the focus is all that is required. The use of the metallurgical lens has a second function. The front element is spring-mounted and slightly recessed, eliminating cracked objectives. This also relieves the tension on the untrained personnel.

(4) Room for a microscope and light source is minimum.

(5) Slight loss of definition occurs resulting in the loss of the shorter tracks and discrimination against lower energy neutrons. The loss is approximately 15–20 per cent in a PoBe spectrum and 30 per cent in a Mock Fission spectrum. However, calibration against a spectrum eliminates this problem and the increased speed still allows us to read one-half again as many films with less personnel training. We read 2.5 mm^2 per film and require $1\frac{1}{2}$ min for each film.

(6) When an experimental film is to be counted to an extremely high accuracy or when the microscope is needed in other work (biology, chemistry, etc.) the bulb is changed to a 100 W and the monocular head replaced with a binocular head. The total time required for this operation is less than 5 min.

(7) The total cost is minimal since the light source cost is approximately 20 per cent of the microscope cost.

G. E. Wilcox

Radiation Laboratory
University of California
Livermore, Calif.

FIG. 1. The projection microscope and its component parts.

Health Physics Pergamon Press 1958. Vol. 1. p. 357. Printed in Northern Ireland

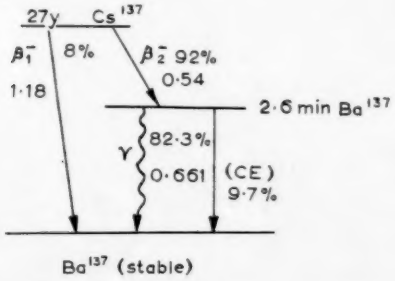
LETTER TO THE EDITOR

Some Comments on the Decay Scheme and Dosimetry of Cs^{137}*

(*Received* 15 *May* 1958)

RECENTLY a shipment of a Cs^{137} source was received where careful measurements showed that the source strength had been evaluated as much as 25 per cent below the ordered quantity. Now this kind of incident could easily occur again, since the reason for the improper evaluation does not lie wholly with the instruments or techniques used so much as with the interpretation of the Cs^{137} decay scheme. Those consulting the U.S. National Bureau of Standards Circular 499, Nuclear Data, and its supplements,[1] and the Table of Isotopes[2] will have to make appropriate calculations in order to determine the γ-energy per disintegration (if one uses the roentgen dosimeter method) or adjust the count rate appropriately to determine the proper disintegration rate, if one is to use $4\pi\beta$ counting. Those who are using Cs^{137} information contained in the Radiological Health Handbook[3] will obtain an inaccurate answer, if the value of r/hr per c as given for specific distances is used.

It is the intent of this letter to allow those health physicists and others who are interested in Cs^{137} dosimetry to take heed in their interpretation of the Cs^{137} decay scheme and data as mentioned above. The following is probably the best information available concerning the decay scheme.



The above scheme is based on the following parameters where α_K is the K conversion coefficient, γ is γ-rays, and K, L and M are internal conversion electrons (CE), $K/(L+M) = \alpha_K/(\alpha_L + \alpha_M)$.

$$\alpha_K = \frac{K}{\gamma} = 0.096 \tag{1}$$

$$\frac{K}{L+M} = 4.6 \tag{2}$$

For each disintegration:

$$K + L + M + \gamma = 1 \tag{3}$$

$$\frac{K}{4.6} = L + M \qquad \text{from (2)}$$

$$K + \frac{K}{4.6} + \gamma = 1 \qquad \text{from (3)}$$

$$1.22K + \gamma = 1$$

$$K = 0.096\gamma \qquad \text{from (1)}$$

$$(1.22)(0.096\gamma) + \gamma = 1$$

$$1.117\gamma = 1$$

$$\gamma = 0.895$$

$$(K + L + M) = 0.105$$

So that $92(0.895) = 82.3$ per cent goes to stable Ba^{137} by γ-emission, and $92(0.105) = 9.7$ per cent by internal conversion electron emission. This would provide a γ-dose rate of 0.31 r/hr per c at 1 m.

If one is counting in a $4\pi\beta$ counter, the correction for counts into disintegrations is $d = c/1.097$ since the betas and conversion electrons are not in coincidence. That is to say, for every 100 disintegrations, one will get 109.7 counts in a $4\pi\beta$ counter.

I sincerely hope that this will help resolve a situation that to my way of thinking needed clarification.

S. BLOCK

University of California
Radiation Laboratory
Livermore, California

* Work was performed under auspices of the U.S. Atomic Energy Commission.

REFERENCES

1. *Nuclear Science Abstracts, New Nuclear Data* Vol. VIII, No. 24B (1954); Vol. XI, No. 6B (1957).
2. HOLLANDER, PERLMAN and SEABORG, *Rev. Mod. Phys.* **25,** No. 2, 469 (1953).
3. S. KINSMAN (Editor), *Radiological Health Handbook* p. 139. U.S. Department of Health, Education and Welfare (1957).

Health Physics Pergamon Press 1958. Vol. 1. pp. 358–361. Printed in Northern Ireland

NEWS

News Editor: SAUL J. HARRIS

Health Physics in the News

THE following items concerning Health Physics have appeared in newspapers, public press or technical journals during the last quarter.

Oak Ridge workers released

During the last quarter all workers exposed to radiation in a criticality incident at the Y-12 plant in Oak Ridge in June 1958 were released from the ORINS hospital [see "Health Physics in the News", *Health Physics* **1,** 207 (1958)]. As an aftermath of the accident, the Oak Ridge Locals of the Oil, Chemical and Atomic Workers stated that they would seek special legislation in the state to protect workers exposed to radiation.

Reactor safety fuse developed

Atomics International Division of North American Aviation announced that it has designed, built and successfully tested a reactor safety fuse for pool type research reactors.

Public liability exemption enacted

An amendment to the Atomic Energy Act of 1954 which became law last quarter indemnifies non-profit educational institutions against public liability in excess of $250,000 in connection with nuclear accidents and exempts non-profit educational institutions from the financial protection requirement of the Anderson–Price Indemnity law. The indemnity applies to institutions operating nuclear facilities under AEC licences issued between 30 August, 1954 and 1 August, 1957.

Mass X-rays questioned

Included in the chapter on "things to think about" of the booklet *Public Exposure to Ionizing Radiations*, published last quarter by the American Public Health Association, was the subject of the use of X-rays in mass screening. The APHA reported that several national medical associations questioned the use of X-rays for mass screening and instead recommended selective mass surveys. They objected to the use of fluoroscopy for mass screening of persons passing through portals or any uses of fluoroscopy that are not under the direct supervision of competently trained personnel.

Weapons fallout data published

The New York Operations Office last quarter published a 4 volume report on *Environmental Contamination from Weapons Tests* which includes all fallout data collected by the laboratory during its monitoring operations. The report is identified as HASL-42, A,B,C,D.

Waste disposal hearings set

The Joint Committee on Atomic Energy, U.S. Congesss, Special Subcommittee on Radiation released its tentative outline for public hearings on "Industrial Radioactive Waste Disposal" to take place in January 1959. Among the subjects to be reviewed are the nature of wastes, waste management operations, waste disposal research and development programs, future estimates and economics of waste disposal, administrative and policy problems and industrial and international problems. A period of seven days is planned for the entire hearings.

Weapons testing advocated

Dr. STAFFORD WARREN, Dean of the Medical School of the University of California, at Los Angeles, last quarter advocated continued nuclear weapons testing to permit man to learn how to live in a radiation contaminated environment. Speaking at a California Optimists Club meeting, Dr. WARREN commented that "solving the problems of living in radioactive contamination after a nuclear war" should receive as much effort as is put into military safeguards. Stating that "this problem of universal contamination is one of the most complex and at the same time the most important medical and public health problems facing the world today", Dr. WARREN added that "we must solve them from actual experience with tracer techniques in the actual field trials of small bomb tests".

Tea shipment tested

A shipment of slightly contaminated (presumably from nuclear fallout) tea was held up in New York City last quarter until the safety of dock workers could be determined. The International Longshoremans' Association requested that the hazards to the unions' members be evaluated by radiation protection

personnel from Columbia University. It was found that there was no hazard to the unloading crew. Subsequently, the tea was studied to determine whether there would be a hazard from radioactive material in the tea if it were sold. The Food and Drug Administration, under whose direction the tests were made, reported that the tea could be sold since the brewed tea would contain far less strontium-90 than the maximum permissible concentration for continuous drinking of water.

Radiation exposure cards delivered

Hanford delivered radiation exposure "report cards" last quarter to more than 8500 atomic workers working in production and laboratory areas of the plant. The General Electric Co., which operates Hanford, announced that these cards report the radiation received by each employee during his entire employment at the plant. Analysis of these records indicated that "in no case has an individual's total accumulated dose of penetrating radiation exceeded or even approached the maximum limits" proposed by NCRP.

U.N. Committee report published

The U.N. 15-member Scientific Committee on the Effects of Atomic radiation completed and published a report of its findings last quarter which, among other things, stated that it is apparent that information concerning the effects of irradiation "is still insufficient.... Any present attempt to evaluate the effects of sources of radiation to which the world population is exposed can produce only tentative estimates with wide margins of uncertainty." The report stated that occupational exposures account for "about 2% of that from natural sources in countries in which occupational exposure is probably the largest". "On the other hand", the report added that "in the countries with extensive medical facilities...the radiation given for medical purposes makes the largest artificial contribution to the irradiation of the population". The U.N. report was endorsed by various international groups and the U.S.A.E.C. The U.N. General Assembly was urged to continue the Scientific Committee and to expand its functions.

Radiation research meeting held

In Burlington, Vermont, at the International Congress on Radiation Research, many papers were presented on the effects of radiation on man and attempts to reduce the damaging effects. In addition, it was reported that among foods which contain natural radioactive materials, in concentrations exceeding fallout in contaminated foods, were Brazil nuts, cereals, shellfish and peanuts. (See, R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD, *Hlth Phys. J.* **1,** 268 (1958); this issue.)

Submarine visit cancelled

Last quarter saw the first refusal to permit a nuclear powered ship to enter a sea port when the Danish government cancelled plans to have the U.S. submarine *Skate* visit that country. The Danish government questioned the advisability of having the reactor-propelled vessel in a populated seaport. The A.E.C. subsequently announced that the British had also required that the U.S. submarine *Nautilus* dock in Portland, England, a seaport which was not densely populated. The A.E.C. also stated that it "is satisfied with the reliability and safety of the nuclear power plants in submarines".

Maritime safety discussed

At the second A.E.C.-Maritime Administration symposium on nuclear ship propulsion, held last quarter in Washington, D.C., it was reported that work on the safety aspects of nuclear ship propulsion has shown that "an analysis of fifty significant ship collisions, including that of the *Andrea Doria—Stockholm*, shows that if the *Savannah* had been struck by the *Stockholm* abreast the reactor, she would have survived without damage to the reactor containment". In a later meeting in Salzburg, Austria, the Maritime Administration stated that nuclear merchant ships will not pollute the seas with radioactive wastes.

Piqua site questioned

Last quarter, the A.E.C's Advisory Committee on Reactor Safeguards tentatively rejected the site proposed for the City of Piqua (Ohio) power reactor as "not suitable". The full reasoning behind the comment was not released.

New York City registration started

Last quarter, the City of New York Department of Health mailed to "more than 30,000" doctors, dentists and other radiation source users in the City registration forms as required by the recently adopted revision to the City's Sanitary Code.

More money for radiation urged

Last quarter the Secretary of the U.S. Department of Health, Education and Welfare advocated that there be more Federal, state and local funds used to protect the public from radiation hazards. In a statement announcing this proposal, the Department also reported that the Food Additive Amendment, enacted by the U.S. Congress in 1958, included a

requirement that any radioactive materials used in food processing must be classed as a food additive, meaning that foods so treated must be proved safe before they can be sold to the public.

Announcements

There will be a session on Radiation Protection sponsored by the American Industrial Hygiene Association at the Industrial Health Conference in Chicago, Illinois, 27 April–1 May, 1959. Persons wishing to present papers should contact Mr. J. A. MARTIN, Pratt and Whitney Aircraft, P.O. Box 611, Middletown, Connecticut.

ICRU Meeting 1958

THE International Commission on Radiological Units and Measurements (ICRU), with members drawn from various countries, has just completed a two-week meeting in Geneva (Sept. 14–26, 1958). Since 1925, the ICRU has been instrumental in (1) the establishment of the units of radiation dose now used on an international basis, (2) the intercomparison of national radiation standards and the evaluation of the results, and (3) the development and evaluation of procedures used for measuring radiation dose both for treatment of cancer and for protection of people against the harmful effects of radiation. The meetings were held with the assistance of the World Health Organization and the Rockefeller Foundation.

Important results achieved at the current meetings included the assessment of a series of international comparisons of the standards of radium used by the national laboratories of six countries. Since the present radium standards were produced in 1934, there has never been a complete international comparison of the values using modern techniques. An evaluation of these comparisons by the ICRU has verified the initial values of the radium standards used by U.S., U.K., Canada, Germany, France, and Austria, and agreement between the measurements shows an accuracy of two-tenths of one percent.

An evaluation was made also of the intercomparisons of eleven artificially produced radioisotopes used in many countries. Some fifteen more are under study. The encouragement of this standardization program by the ICRU began five years ago, at which time international agreement was very poor and only one or two isotope standards were available internationally. Evaluation of the current situation shows that present agreement between isotope standards over the world is in most cases two percent. Such agreement permits the comparison of medical and experimental results in the various countries to a degree not possible a few years ago.

An evaluation was made also of the status of standards used for X-ray dosage and measurements between the national laboratories of five countries. The program of standards comparisons was initiated by the ICRU in 1953 at which time there was considerable uncertainty as to the status of the situation. In the first stages of this program, the standards themselves were moved between countries, but because of the impracticality of this, the ICRU in 1956 recommended the development of a small secondary standard which could be readily transported between countries. Such a standard was produced by the National Bureau of Standards (U.S.A) with the financial assistance of U.N.E.S.C.O. and W.H.O. Some initial international comparisons with this instrument were reported and plans laid for extending these to include other countries such as Italy, Australia, New Zealand, France, Holland, Canada, and U.S.S.R. Evaluation of the results of the intercomparisons made thus far indicate that the X-ray standards for radiation dose are in agreement within half of 1 per cent. Such standards are relied on to provide not only comparable doses of radiation in medical therapy and diagnosis but also to calibrate instruments used for measuring radiation in the neighborhood of atomic energy and other radiation-producing installations.

The ICRU also reached agreement on the specification of radiation dose to be used in clinical and protection measurements for X-rays and radioactive materials used either external to the body or within the body for therapeutic purposes.

The Commission agreed to institute a long-range program on the whole system of units for future radiation measurements. It is not expected that the results of this study will be available for several years.

The Commission also called together, for a full-day meeting, representatives of the United Nations specialized agencies and non-governmental organizations having concern with problems of radiation protection, units, and measurements. This meeting provided an unique forum for the exchange of views and information relating to this very complex problem and laid the groundwork for better future cooperation between all the groups involved.

The final report of the International Commission on Radiological Units and Measurements will not be available for several months. It is expected that the detailed results of this meeting will be reported publicly for the first time at the International Congress of Radiology in Munich in July, 1959.

H. O. WYCOFF

Plan for Monitoring Bone-Seeking Radioactive Contamination in Milk

THE Division of Laboratories and Research of the New York State Department of Health is inaugurating a statewide plan for monitoring bone-seeking radioactive contamination in milk. This contamination is primarily due to the ingestion and inhalation of fallout fission products by cows. Milk and the bone-seeking elements were chosen as the basis for the investigation because milk is the single most important source of dietary calcium of people in the western hemisphere. It may be a particularly important source of radiation exposure to the rapidly developing embryo, fetus, or newborn child because metabolically the bone-seeking fission products act like calcium.

The hazard to populations from fission product contaminated milk is presently considered to be largely somatic. This somatic hazard, in turn, is considered to be almost entirely due to the irradiation of the bone and bone marrow by the bone-seeking radioisotopes. Therefore, the New York State Department of Health is searching for the following information from this project:

(1) What are the total bone-seeking radioisotopes in the local fluid mild produced in New York State?

(2) What is the total dose to bone and bone marrow due to this activity in milk?

A method for the determination of the bone-seeking radioisotopes has been developed in the laboratory. A 100–500 g sample of fluid milk is extracted with dilute nitric acid. The calcium and calcium-like atoms are then precipitated as oxalates and ignited to oxides. In a few hours the bone-seeking radioisotopes may be concentrated and counted. The oxide ash has been successfully counted for total beta activity by liquid scintillation techniques, anti-coincidence low background beta counting, and proportional counting. By methods of differential absorption and/or decay studies, estimates of the dose to bone and bone marrow may be calculated for each sample. Subsequent study on tissue uptake of the various bone-seeking elements in comparison to calcium, skeletal distribution of the bone-seeking radioisotopes taken into the body by various routes, and radioisotopic composition in the milk will help to better determine the estimates of the dose to critical organs.

N. IRVING SAX
ROBERT J. SHERER

Proposed Health Physics Society in Europe

SEVERAL health physicists working in European countries have thought for some time that it would be desirable to form a branch of the Health Physics Society in Europe. During the second U.N. International Conference on the Peaceful Uses of Atomic Energy held in Geneva in September, the opportunity was taken to hold a meeting at which the possibility of this could be discussed. By the courtesy of Dr. C. J. BAKKER, Director of C.E.R.N., the meeting was held in that Establishment. The meeting was called by an "Inviting Committee" consisting of W. G. MARLEY (U.K.), B. M. WHEATLEY (C.E.R.N.), P. COURVOISIER and W. HUNZINGER (Switzerland), and about 70 or 80 persons concerned with health physics or radiation protection work in many European countries attended.

Following the opening remarks by Dr. MARLEY, who acted as Chairman, the meeting was privileged to be addressed by Dr. LAURISTON S. TAYLOR, President of the Health Physics Society. Dr. TAYLOR gave an account of the history of the Health Physics Society from its earliest beginnings to the most recent developments which now embraces Local Chapters and Regional Sections. Dr. TAYLOR strongly supported the move in Europe for the establishment of a European Section of the Health Physics Society.

In the ensuing discussion it was envisaged that a European branch of the Health Physics Society could readily be formed, that it would be substantially self-supporting financially, that it would obtain the same favourable terms for copies of the *Health Physics Journal*, and that it would participate as a Regional Section of the parent Society.

The meeting went on to discuss an outline constitution of a European Health Physics Society and following extensive discussion of the objectives and functioning of such a Society, the meeting appointed a Working Committee to revise the draft Constitution and establish a European Section of the Health Physics Society. The meeting elected the following persons to serve on the *Working Committee*

W. G. MARLEY (Chairman)	United Kingdom
H. JAMMET (Vice-Chairman)	France
W. HUNZINGER (Secretary)	Switzerland
B. BELLION	Italy
R. BOULENGER	Belgium
H. GRAUL	Germany
S. LINDHE	Sweden
J. J. ZAKOVSKY	Austria

The Working Committee will circulate the revised Constitution to participants of the meeting for a letter ballot.

All persons who are engaged in health physics or radiation protection work in European countries are invited to get into touch with the Secretary of the Working Committee, Dr. W. HUNZINGER, Reaktor Ltd., Würenlingen, Aargau, Switzerland, requesting particulars of the proposed Society and indicating whether they wish to join.

W. G. MARLEY

PAPERS TO BE PUBLISHED IN FUTURE ISSUES

A. Walton, R. Kologrivov and J. R. Kulp: The concentration and distribution of radium in the normal human skeleton.

Ann T. Nelms and J. W. Cooper: U^{235} fission product decay spectra at various times after fission.

Betsy J. Stover: Metabolism of radioactive isotopes which deposit mainly in the skeleton.

H. P. Yockey: Radiation aging and its relation to the principles of Health Physics.

C. F. MacGowan: Comments on the labor relations aspect of nuclear safety.

G. M. Dunning: External β-doses from radioactive fallout.

W. H. Langham and E. C. Anderson: Cs^{137} biospheric contamination from nuclear weapons tests.

D. Callihan and J. Thomas: Accidental radiation excursion at the Oak Ridge Y-12 Plant—I. Description and physics of the accident.

J. D. McLendon: Accidental radiation excursion at the Oak Ridge Y-12 Plant—II. Health Physics aspects of the accident.

E. L. Geiger: Radioassay of uranium and plutonium in vegetation, soil and water.

R. H. Ritchie and G. S. Hurst: Penetration of weapons radiation: application to the Hiroshima–Nakasaki studies.

ERRATA

p. 126: Item (c) should read: The values of MPC and the resulting q must be such that the quarterly RBE dose to the gonads or to the total body will not exceed 3 rems, and the average dose to age N will not exceed $5(N - 18)$ rems.

p. 131: τ in equation (2) should read t.

pp. 145–150: The dose figures cited are understood to be for whole body exposure to X- or γ-radiation.

p. 146: Line 4, should read: With respect to acute whole body exposures . . .

p. 175: Neutron flux is given as n/cm. This should be n/cm^2.

Health Physics Pergamon Press 1959. Vol. 1, pp. 363–372. Printed in Northern Ireland

ACCIDENTAL RADIATION EXCURSION AT THE OAK RIDGE Y-12 PLANT—I

DESCRIPTION AND PHYSICS OF THE ACCIDENT

DIXON CALLIHAN and JOSEPH T. THOMAS
Oak Ridge National Laboratory,* Oak Ridge, Tennessee

(*Received* 13 *November* 1958)

Abstract—An aqueous solution of enriched uranium inadvertently flowed into a 55 gal drum in a process area in Oak Ridge in June 1958, establishing a prompt-critical neutron chain reaction in which about 10^{18} fissions occurred before the system finally became subcritical by the addition of water. The solution contained about 2.5 kg of U^{235}. Records of the radiation field show the power excursion to have continued about 20 min during which the reaction oscillated a number of times. This paper describes the accident and presents a reactor-physics analysis yielding reactivities in an unperturbed system as great as 1.3 per cent which were above zero for a time consistent with observations. A plausible sequence of events during the excursion is enumerated. The emergency and health physics procedures and the medical observations of exposed personnel will be given in subsequent papers of this series.

INTRODUCTION

Estimates have been made of the expected consequences to personnel and equipment of an uncontrolled neutron chain reaction outside a nuclear reactor ever since the inception of chemical and metallurgical processes with fissionable materials. These estimates have been primarily extrapolations of the experiences with nuclear accidents in experimental critical assemblies[1] although at least one analytical treatment of an assumed set of conditions has appeared in the literature.[2] The accident that occurred in one of the salvage areas in Oak Ridge in June 1958 is the first recorded radiation excursion in a process area and resulted from the inadvertent accumulation of an aqueous enriched-uranium solution in a process vessel. This occurrence, which has been reported fully elsewhere,[3] was an important experience in applied health physics since the primary and immediate concern was an evaluation of personnel exposure to penetrating radiation and the establishment of necessary remedial measures. A study of the health physics problems and of the medical findings will be presented in subsequent papers in a series of which this is the first. The cause of the accident and an analysis of it from a reactor physics viewpoint, both of which are discussed in this paper, are of less overall importance except as they aid in establishing the radiation levels to which personnel may have been exposed. Therefore, only a brief description of the cause and a review of what is believed to be the sequence of events leading first to the critical condition within the solution and its final return to subcritical are presented here. One plausible reactor physics analysis of the power excursion is also considered.

DESCRIPTION AND ANALYSIS

The nuclear accident occurred in an area in which salvable enriched uranium (~90% U^{235}) is recovered from various materials by physicochemical methods in a complex of equipment. This recovery process was being remodeled at the time, and the situation was

* Operated for the U.S. Atomic Energy Commission by Union Carbide Nuclear Company.

further aggravated by an inventory then in progress. The latter required disassembly, cleaning, reassembly and leak testing of certain pieces of equipment, particularly several long 5 in. diameter pipes used for storage of aqueous solutions of U^{235} whose shape and dimensions prevent the establishment of nuclear chain reactions. The leak testing operation consisted in filling the pipes with water from a 55 gal stainless steel drum (approximately 22 in. in diameter and 33 in. high), observing the joints in the system and subsequently draining the water back into the same drum. These inventory procedures extended over several days and it was not possible or economic to schedule them concurrently throughout the several stages of the salvage train. As a matter of fact, operations had been re-established in the step immediately ahead of the scene of the accident some hours prior to its occurrence and a part of the bank of storage pipes in question had been leak tested a few days previously. As a consequence of both this time factor and irregularities in the function and operation of some valves, a quantity of solution was inadvertently transferred from the area already in operation into the one still undergoing leak testing. Further, this transfer was into one of the 5 in. diameter pipes which had already been tested. It has been established subsequently that the flow pattern from these pipes into the drum intended to receive the leak-test water was such that the accumulated solution preceded the water. The dimensions of the drum and the concentration of the solution permitted the system to become critical. The reaction was terminated sometime later by the inflow of a relatively larger volume of the water believed to have been the only liquid present in the system. A photograph of the drum in the position occupied during the accident is shown in Fig. 1.

A specification of the manner and rate of establishment of the neutron chain reaction system, the determination of the time which elapsed between its first becoming critical and its final return to subcritical together with the power pattern within this interval, and the mechanism by which the nuclear reaction was ultimately terminated would constitute a minimal description of the event. Although the process of transfer of liquid from one vessel to another is fundamentally simple, it is corrrect to infer from the above description of the present operation that many of the details of this transfer are not known even after some careful attempts at reconstruction with non-reactive solutions. It should be pointed out, parenthetically, that although the liquid transfer can certainly typify chemical operations in which accidents of this kind may be expected to occur, it is not believed that this same series of events would ever again ensue, thereby duplicating the consequence. For these reasons any value

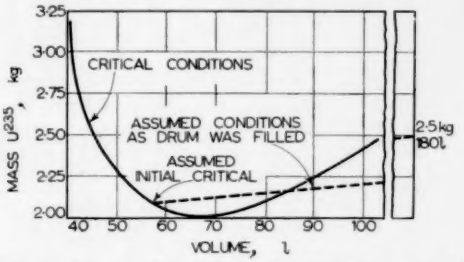


FIG. 2. Mass–volume relations in 55 gal drum during radiation excursion.

of a detailed analysis to the field of reactor physics is doubtful. Although there is no evidence of any basically unexpected physical phenomenon, a complete analytical description of the critical event, agreeing with the observations, would be gratifying and would satisfy the scientific curiosity of many readers. Unfortunately such a description is not possible. A great many observations have been combined to present here a qualitative discussion of the course of events.

A chemical analysis showed 50 g of U^{235} per l. to be the most concentrated solution available for transfer to the drum and 2.5 kg of U^{235} as the mass transferred. A plot, Fig. 2, of a short extrapolation of measured critical dimensions of $U^{235}O_2F_2$ solutions ($\sim$90% U^{235}) gives critical masses as a function of critical volumes in a 21.75 in. diameter unreflected steel cylinder. It is seen that the above values of the chemical concentration and mass set 7.6 and 17.2 in. as the lower and upper limits on the critical height. Since both the sequence of

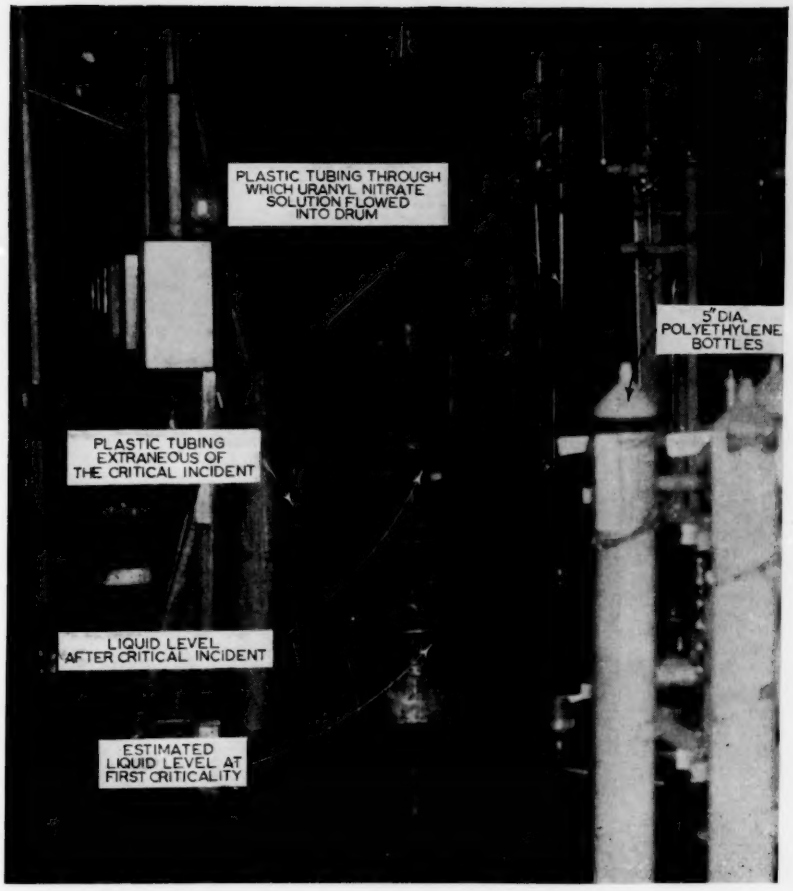

PLASTIC TUBING THROUGH WHICH URANYL NITRATE SOLUTION FLOWED INTO DRUM
5" DIA. POLYETHYLENE BOTTLES
PLASTIC TUBING EXTRANEOUS OF THE CRITICAL INCIDENT
LIQUID LEVEL AFTER CRITICAL INCIDENT
ESTIMATED LIQUID LEVEL AT FIRST CRITICALITY

FIG. 1. Drum (55 gal) in which the critical incident occurred.*

* *Correction:* The wording on the figure "Plastic tubing extraneous of the critical incident", should read "Plastic tubing extraneous to the critical incident".

Table 1. *Calculated reactivity during radiation excursion*

Time (min)	Solution				Critical Mass† U^{235} (kg)	Reactivity $\times 10^3$
	Height		Volume (l.)	Mass U^{235} (kg)		
	(cm)	(in.)				
0.0*	23.45	9.23	56.2	2.10	2.10	0
1.8	25.07	9.87	60.1	2.11	2.04	7.5
5.4	27.12	10.68	65.0	2.12	2.02	11.4
9.0	29.20	11.50	70.0	2.13	2.03	12.4
12.5	31.29	12.32	75.0	2.15	2.07	11.8
15.3	32.82	12.92	78.7	2.16	2.15	7.5
20.0	35.67	14.04	85.5	2.18	2.18	0

* The drum was delayed critical at this point.
† This mass in the volume shown at the left will be critical.

valving operations postulated and the data from the hydraulic reconstruction experiments stipulate some dilution of the original solution as it flowed into the drum, a volume of 56.2 l. containing 2.10 kg of U^{235} standing at a height of 23.45 cm (9.23 in.) is selected as the initial delayed critical configuration. This selection is justified by these factors: the reactor analysis which has been made, based on these initial conditions, yields a time interval consistent with what is believed to be the observed duration of the excursion, and the assumed critical height agrees both with the liquid level estimated in the drum by the individual standing nearby at the time of the first indication of a reaction and with the distribution of induced activity in the walls of the drum described below. If it is assumed that the concentration of the solution subsequently added to the drum was uniform and that the volume in the drum reached 180 l. when the entire 2.5 kg of U^{235} had been transferred, the mass–volume relation in the drum is described by the straight line on the plot. It is recognized that this simplifying assumption is somewhat unrealistic and overestimates the time interval between delayed and prompt critical. It does, however, provide a lower limit to the reactivity available as a function of time. The details of the analysis are given in the Appendix and the results are in Table 1. The reactivity as a function of the solution height in the drum and of the time after delayed critical is shown in Fig. 3. The time scale was derived from some of the post-accident hydraulic measurements, particularly the rate of flow of liquid into the drum. The duration of the excursion, by this analysis, was 20 min. The effects of the neutron absorption by the nitrogen

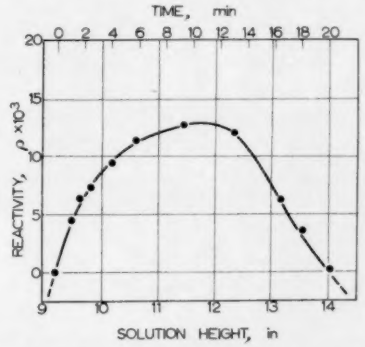


FIG. 3. Calculated reactivity in drum during radiation excursion.

and of the neutron reflection by the concrete floor, located approximately 3 in below the drum, were somewhat compensating and have been neglected. The bases for, and the results of, the above analysis are also not inconsistent with the following additional significant observations.

Records from radiation monitors

During the excursion a radiation detection instrument, sensitive to both neutrons and γ-rays, was operating some 1400 ft distant and

cross the wind from the scene of the accident. Fig. 4 is a reproduction of its output recorded during that time. The following discussion is based on a $\times 7$ enlargement of this record, although most of the points are discernible on the reproductions shown here. The radiation intensity is observed first to have increased extremely rapidly from point (a), driving the pen off scale, to have decreased to point (b), and then to have repeated the sequence to point (c), all in about 15 sec as determined by the chart drive speed. During the next interval, the signal oscillated an indeterminate number of times, finally decreasing to about five times background 2.8 min after the first rise in level. The upper and lower limits of some of these pulses, discernible on the enlarged trace, are indicated by *u* and *l*, respectively. This (average) high-intensity field was then followed by a slowly decreasing level of some 18 min duration, again characterized by pulses. One peak, at 61 on the scale, is separated inordinately in time from adjacent portions of the trace and may be due to a momentary peculiarity of this detector, particularly since it is not readily identified on the charts from either of the air monitors referred to below. Although this neutron detector is equipped with two sensitivity ranges (25 and 125 mr/hr, full scale, respectively), it is believed to have remained on the more sensitive scale during the entire period, discounting the inference that some of the discontinuities are due to scale changes.

An enlargement, Fig. 5, of a section of the chart reproduced in Fig. 4 illustrates qualitatively the power pulses which occurred during the extended period of relatively lower activity.

The overall duration of the excursion is shown by this trace to have been 21 min. The absence of a strong neutron field within the drum as it initially became critical may mean that the critical height was reached prior to the initial energy release, that is, even though the system was critical, it did not manifest itself until it was "triggered" at a low power level, in a statistical manner, by ambient neutrons. This dormant period may have been a few tens of seconds, well within the accuracy of the above estimate.

Two additional radiation monitoring instruments were operating during the time of interest, both being air samplers which detect the γ-radiation from particulates collected on a filter surrounding a Geiger tube. Figs. 6 and 7 are copies of the records from these instruments. Each chart shows the direct radiation from the excursion and, subsequently, the arrival of the air-borne activity. The differences in the interval between the detection of these two activities at the two locations, about 12 min and 48 min, respectively, can be qualitatively correlated with the recorded wind direction at that time. The former was down-wind from the accident. The latter was located in an area adjacent to the site of the detector discussed above, i.e. cross wind from the accident, so the delay in the arrival of air-borne activity is expected to be comparable and equal to about $\frac{3}{4}$ hr. This observation is presented as evidence favoring the interpretation of the extended, low-level activity shown in Figs. 4 and 5, being direct radiation. In addition, of course, Fig. 5 does not typify a radioactivity decay curve. No other quantitative interpretation is made of Figs. 6 and 7.

There are a number of undocumented observations made with portable radiation detection instruments in the vicinity of the accident to the effect that the radiation level remained constant for times of from 5 to 15 min, which is at least supporting evidence that the source of radiation was extended in time.

Analysis of induced activity in the drum wall

Activity was induced by neutrons in the components of the stainless steel of which the drum was constructed. Analyses of these activities yield at least relative values of the neutron exposure and, hence, of the neutron flux at various elevations along the side of the drum. The fast neutron measure was derived from the activity of Co^{58} arising in the $Ni^{58}(n, p)$ Co^{58} reaction. The thermal neutrons were evaluated from the Cr^{51} activity from the Cr^{50} $(n, \gamma)Cr^{51}$ reaction. (An analysis of the steel showed that it contained 17.99 and 11.84 wt % chromium and nickel, respectively.) The activation data are recorded in Table 2 and are

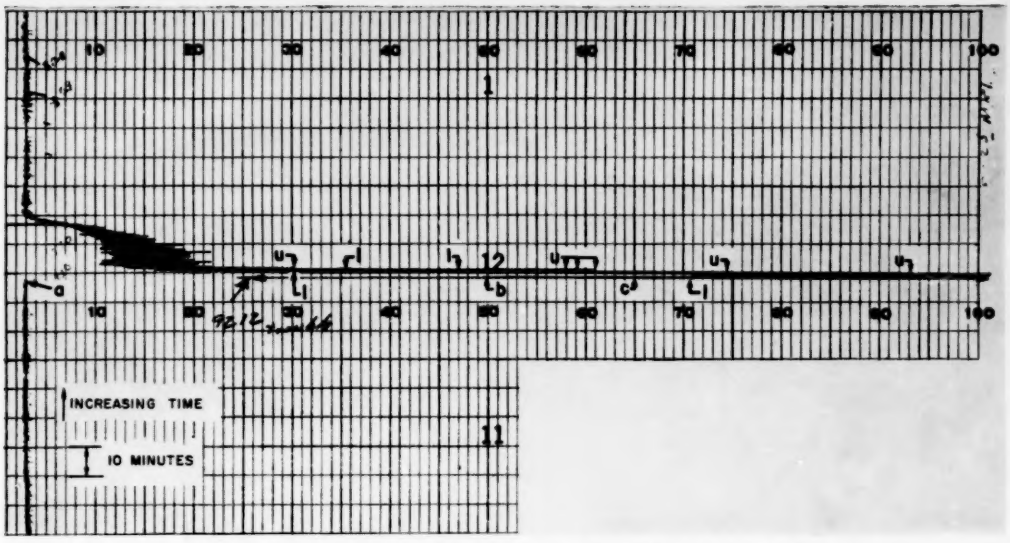


FIG. 4. Recorder chart from neutron detector showing direct radiation.

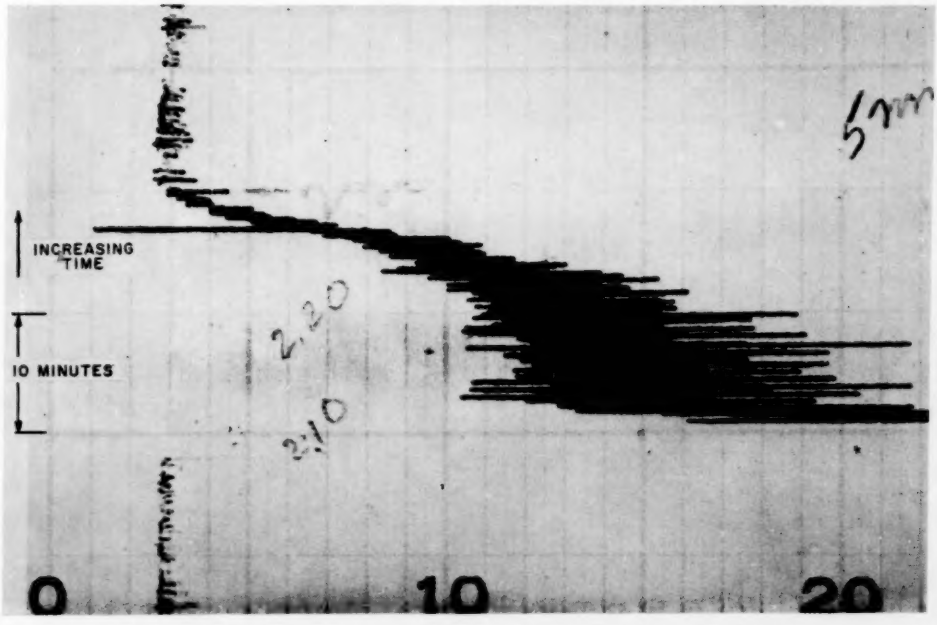


FIG. 5. Section of recorder chart of Fig. 4. Enlarged approximately four times.

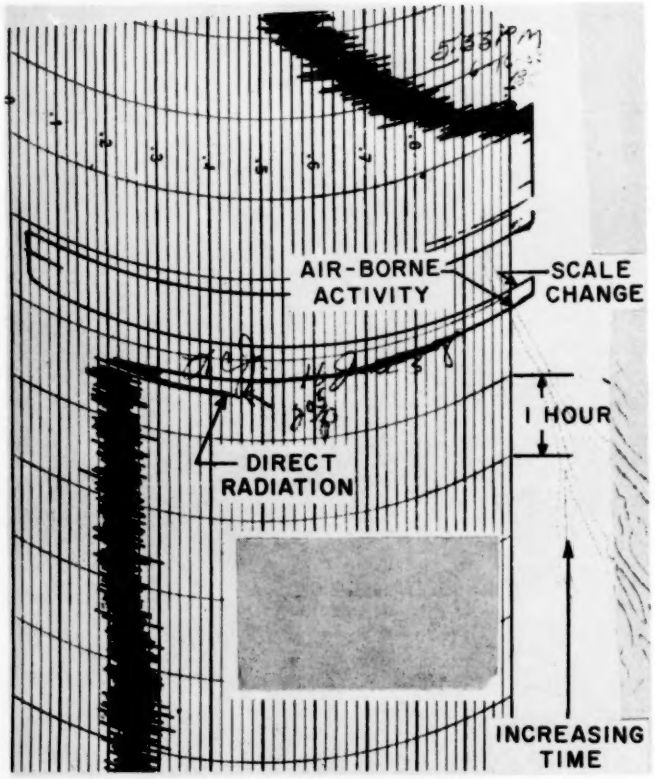


FIG. 6. Recorder chart from air sampler located down wind from accident.

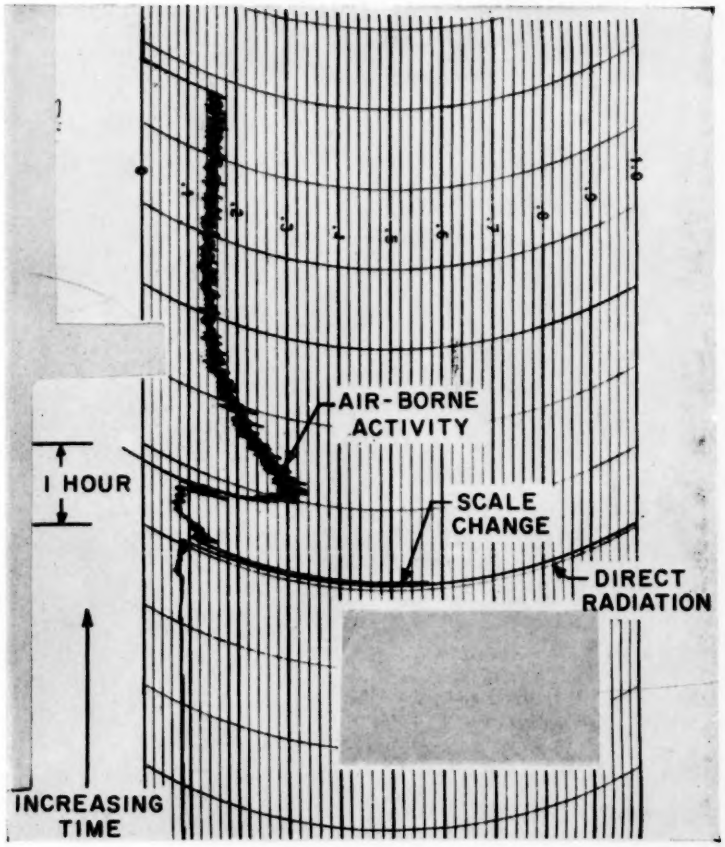


FIG. 7. Recorder from air sampler located crosswind from the accident.

Table 2. Relative activities of stainless steel samples from drum

Height from bottom of drum (in.)	Activation* (arbitrary units)	
	Thermal neutron	Fast neutron
$15\frac{1}{2}$	1.0	4.8
$13\frac{1}{2}$	1.1	4.9
$11\frac{1}{2}$	1.2	6.1
$9\frac{1}{2}$	2.1	9.1
$7\frac{1}{2}$	2.9	13
$5\frac{1}{2}$	3.8	14
$3\frac{1}{2}$	3.9	14
$1\frac{1}{2}$	3.8	11
center bottom	18	28

* The values were obtained by γ-ray spectrometry; radiochemical analysis of three typical samples gave fast-neutron activations from 5 to 15 per cent lower.

plotted in Fig. 8. The results from additional samples from peripheral locations at three elevations show no significant asymmetry in the flux pattern in horizontal planes.

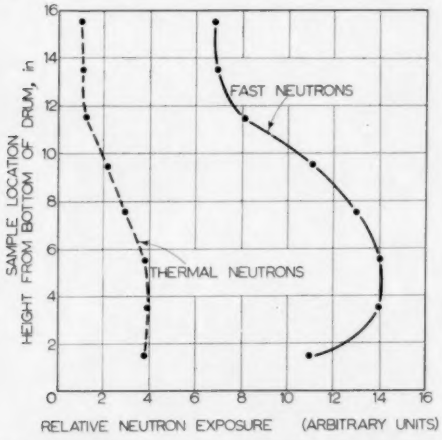


FIG. 8. Relative neutron exposure of samples from side of stainless steel drum.

It is interesting to note that the maximum activation occurred between 3 and 5 in. from the bottom, and that there is some evidence of asymmetry in the thermal neutron distribution, implying an effect of the stainless steel covered concrete floor as a reflector. If the peak activity is associated with some weighted center of reactivity of the supercritical system, an effective reactor height of 10 in. is not inconsistent with the assumptions in the above analysis. No estimate of the energy in the excursion has been made from these values of the steel exposure.

Chemical and radiochemical analyses; energy release

The number of fissions which occurred during the power excursion, and hence the energy release, has been determined from radiochemical analyses of samples of the activated uranium solution. A sample of limited size was taken from the top of the liquid in the drum about 8 hr after the accident. Since this sample may not have been representative of the entire volume of the solution, a pair of samples was taken about 1 month later from the well-homogenized solution as it was then stored in shielded containers. It must be pointed out that some dilution of the solution occurred upon transfer from the drum to the storage containers which accounts for differences observed in the specific activities and the solution volumes. This, of course, in no way invalidates the method, provided the volume is measured at the time of sampling. From the concentration of appropriate fission products (obtained by measuring their characteristic radiation),

Table 3. Fission densities of solution determined by radiochemical analyses of samples

Uranium concentration:*
sample I, taken 8 hr after accident: 14.0 g of U^{235} per l.
sample II, taken 1 month after accident: 9.6 g of U^{235} per l.
sample III, taken 1 month after accident: 9.6 g of U^{235} per l.

Nuclide measured	Method†	Fission density (fissions/ml)		
		Sample I	Sample II	Sample III
Mo^{99}	β count	7.7×10^{12}		
Ba^{140}	β count	6.0×10^{12}	2.8×10^{12}	3.0×10^{12}
	γS‡	6.5×10^{12}		
La^{140}	γS§	4.6×10^{12}		
Ba^{139}	γS	2.2×10^{12}		
Ce^{141}	γS		5.8×10^{12}	5.6×10^{12}
Ce^{144}	β count**		4.1×10^{12}	4.0×10^{12}
Zr^{95}	γ count		3.5×10^{12}	3.6×10^{12}
Cs^{137}	γS		0.6×10^{12}	0.6×10^{12}
Sr^{89}	β count		0.5×10^{12}	0.5×10^{12}
Weighted "best value"		7×10^{12}	5×10^{12}	5×10^{12}

* From chemical analyses.

† The activities were measured by β- or γ-ray counting (β count or γ count) or by scintillation spectrometry (γS).

‡ Assuming 21.5 per cent of the disintegrations yield 0.54 MeV γ-rays, which is based on unpublished data of LYON, Oak Ridge National Laboratory.

§ After several hours growth in separated barium.

** The Pr^{144} beta particles were measured through an aluminum absorber (104 mg/cm²) used to reduce the Ce beta particles. The presence of this absorber necessitated a 48 per cent correction to obtain the Pr^{144} beta yield.

Table 4. Estimates of energy release during accident

	Based on sample I	Based on samples II and III
Volume of solution yielding sample (l.)	180	252.8
Mass of U^{235} in total volume (kg)	2.5	2.4
Total number of fissions	1.3×10^{18}	1.3×10^{18}
Energy release	2.6×10^{20} MeV = 11 kWh	

together with their decay constants and fission yields and the elapsed time between the excursion and the analysis, the number of fissions which occurred per unit volume of the solution was obtained. All of the analytical results and a weighted "best value" of the energy released in the excursion, 2.6×10^{20} MeV from 1.3×10^{18} fissions, are given in Tables 3 and 4.

It will be noted that large discrepancies exist in the data of Table 3. A partial explanation lies in the existence of noble-gas precursors of most of the nuclides measured in the analysis. A list of these precursors is given in Table 5.

Table 5. Properties of fission product nuclides measured in analysis of solution

Nuclide	Fission yield fraction[4]	Gas precursor
Sr^{89}	0.048	3.2 min Kr
Zr^{95}	0.064	"short" Kr
Mo^{99}	0.062	—
Cs^{137}	0.059	3.9 min Xe
Ba^{139}	0.063	41 sec Xe
Ba^{140}	0.061	16 sec Xe
Ce^{141}	0.060	1.7 sec Xe
Ce^{144}	0.061	~1 sec Xe

Gases of longer half-lives obviously have higher escape probabilities from the liquid than those of short half-lives. Further confirmation of this explanation is obtained from observations on samples of solutions in which the fission concentrations have varied; the difference between the fission concentration values derived from Ba^{139} and Mo^{99} increases with increasing concentration, i.e. increasing heat output. The apparently low values of the fission concentration in the latter sample, based on Zr^{95} and Ba^{140}, reported in Table 3, may be explained by the well-known hydrolytic behavior of zirconium and possible similar loss of barium due to traces of sulfate (in addition to the loss of 16 sec Xe^{140}). Disagreements between values from Ce^{141} and Ce^{144} have not been explained.

Hydraulic reconstruction experiments

Considerable effort was expended in attempts to reconstruct the flow patterns of the several volumes of liquids as they were added separately to a somewhat complex system of piping, partly mixed therein, and finally drained into the 55 gal drum in a stream of variable uranium concentration. An aqueous solution of cadmium nitrate, adjusted in concentration to approximate the fluid properties of the mislocated uranium solution, together with the volume of water believed appropriate, were used in these tests. Flow rates into the drum were measured and frequent samples were obtained both from the top of the liquid in the drum and from the line as the drum was filled. Although, in principle, the analyses of these samples allow an estimation of the uranium inventory and concentration in the drum as a function of time, it is not certain they are truly representative of the conditions in the drum at the time of the accident. This uncertainty may be due, for example, to irreproducible mixing conditions, particularly since the first emission of nuclear energy caused at least local turbulence. The fill rate was used in the above reactivity analysis, but it has not been possible to correlate the time–uranium inventory data with the uranium concentrations required for criticality.

General observations

There are two additional observations which should be recorded for consideration. One of them is the absence of a strong ambient neutron field at the scene of the accident (the most likely source being the $O(\alpha, n)Ne$ reaction between the U^{234} α-particles and the oxygen in the water) and, as a consequence, the system may have been above delayed critical before the power level increased from zero.

The second observation is that there was no evidence of the rapid production of large quantities of gas or vapor. There was, for example, no liquid on the floor under or adjacent to the drum, nor was there an inordinate amount of localized fission product contamination on the fill tube (see Fig. 1) except where it was in contact with the liquid. The nature of the process in the area precluded any meaningful α-particle contamination survey for dispersed uranium. These observations minimize any assumption of vigorous boiling of

the solution. There is no clear explanation of why the solution was not dispersed outside the drum, although speculation can relate the violence of the turbulence to the rate and mode of the approach to critical, to the characteristics of the first power surge, and possibly, to the geometry of the vessel. Comparison of experiences with other critical accidents[1] with solutions shows that large as well as insignificant discharges of liquid have been observed in events with the same energy release.

DISCUSSION

An attempt has been made in the preceding paragraphs to record and interpret a rather wide variety of observations made in connection with the radiation accident. It is believed, unquestionably, that sufficient enriched uranium solution was added to a 55 gal drum to become critical, that the concomitant energy release occurred during an interval of a few minutes in which the effective reactivity and the power level oscillated a number of times, and that the chain reaction was ultimately stopped by the addition of water to the solution (since, very fortunately the valve, through which the solution was admitted, was left open as personnel evacuated the area). The quantity of uranium involved and the energy developed in the reaction are moderately well known; the uncertainty in the duration of the excursion and the fluctuation in the reactivity have not allowed an evaluation of the peak power. The potential personnel hazard from the ionizing radiation generated in the observed number of fissions is developed elsewhere in this series of papers and is compared with the exposures experienced by employees in the vicinity of the accident.

As pointed out earlier, it is impossible to reconstruct the reactivity–time pattern and there are, no doubt, several combinations of events which can account for the observations. It is intended to outline very briefly here one possible sequence.

With reference to the power-level relation, indicated by the radiation monitor record described in Fig. 4, the following sequence of conditions is suggested. In the absence of a source of neutrons, this system was prompt critical before any energy was emitted. Once started, however, the power level rose quite rapidly to a high value. The energy from these fissions produced gases by dissociation,[5] reducing the density and driving the solution subcritical. Exit of these gas bubbles once more made the system prompt critical and, with the delayed neutrons as a source, the power level again rose. This cycling persisted for an estimated 2.8 min, during which, of course, the temperature of the solution increased. Boiling* finally ensued, causing a sharp decrease in density and a concomitant return to subcritical indicated by the decrease in the instrument deflection to about scale reading "20," (Fig. 4). Following this steep descent, the system settled into an equilibrium condition somewhere in the delayed critical range where it was controlled for about 18 min by vapor formation and, to a lesser extent, by decomposition gases. The system remained delayed critical until the inflow of water reduced the concentration to a final subcritical value.

In previous experiences with accidental critical assemblies,[1] which have been limited to a single burst by some reactor shutdown mechanism, the energy release has been from 10^{16} to 10^{17} fissions, a not unreasonable estimate of the first of the several pulses in this case.

It is appropriate to consider, briefly, other courses the reaction may have taken and the consequences which could have resulted. For example, one shutdown mechanism for a supercritical solution, alternate to a dilution, is the removal of sufficient water to increase the chemical concentration beyond that which will support a nuclear chain reaction under the other existing conditions. This removal would be by dissociation and vaporization. In this particular instance, the above analysis shows (Fig. 2) the limiting concentration to be about 54 g of U^{235} per l. with a total of 2.5 kg of U^{235}, a value, incidentally, not much different from that of the original solution. Had no

* The permanent deformation of a polyethylene liner, present in the drum during the excursion, into the convolutions of the drum is evidence that the temperature of the solution at least approached the boiling point. The energy release obtained from the fission product analyses was adequate to boil the solution.

water been added in the operation, the excursion might not have been as severe as the one experienced.

Another shutdown mechanism is a dispersal of the fissionable material, the causes of which are difficult to predict from past experience.

It is believed that the incident described here is a point of departure for predicting the causes and effects of possible future accidents. It does not set an upper limit to the consequence to be expected for, as pointed out above, there were associated with it a number of unique, fortunate circumstances which reduced the problem significantly. A study of this type of accident has been made,[2] which is supported in part by the findings reported here, and which, in the absence of externally applied shutdown mechanisms, predicts much more severe results.

Acknowledgments—The co-operation of many persons in making the study of the radiation excursion is gratefully acknowledged. Particular mention must be made of helpful discussions with F. S. PATTON and G. R. JASNY of the Y-12 Plant in Oak Ridge and with J. R. KNIGHT and J. C. BAILEY of the Oak Ridge Gaseous Diffusion Plant. The radiochemical analyses were made under the direction of S. A. REYNOLDS and E. I. WYATT of the Analytical Chemistry Division of the Oak Ridge National Laboratory.

APPENDIX

Method of calculating reactivity

The critical mass in an unreflected 21.75 in. diameter stainless steel cylinder was determined as a function of the critical volume by equating its geometric buckling to that of a similar cylinder 20 in. in diameter for which the critical parameters are known. Once the variation of critical mass with critical volume of the larger cylinder is known, an initial critical point on the curve, commensurate with facts observed after the excursion, is chosen. This point, *A*, in Fig. 9, represents 2.1 kg of U^{235} in 56.2 l. of solution. The subsequent masses and volumes, as additional solution enters the drum, are represented by line *AB*, assuming that the concentration of the incoming solution remains constant. It is further assumed that the final contents are 2.5 kg of U^{235} in 180 l.

In a two neutron-energy group analysis, the effective reactor multiplication factor, k, of critical and near critical assemblies is related

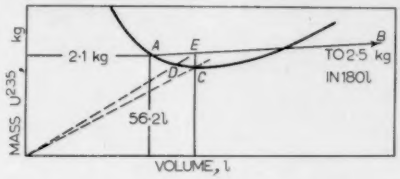


FIG. 9. Schematic diagram of critical mass in an unreflected 21.75 in. diameter stainless steel cylinder as a function of critical volume.

to the material and geometric properties of the assembly by

$$k = \frac{\eta f}{(1 + L^2B^2)(1 + \tau B^2)}$$

where η = number of fission neutrons produced per neutron absorbed by U^{235}

f = thermal neutron utilization

L^2 = square of the thermal diffusion length

B^2 = geometric buckling of the reactor

τ = neutron age

Along the critical curve in Fig. 9, the equation has the value unity, of course, and the geometric and material buckling are equal.

As the cylinder continues to fill, the mass and volume increase to point E which describes a different (supercritical) combination of geometry and material. The nuclear properties of the latter are the same as those of the solution critical at point D, since a line through the origin represents a particular chemical concentration, and the values of ηf at D and E are, therefore, equal. Since the geometric buckling at conditions C and E are the same and L^2 and τ are essentially constant over this concentration range, the multiplication constant at E is given by:

$$k_E = \frac{(\eta f)_D}{(\eta f)_C}$$

REFERENCES

1. J. T. Thomas and A. D. Callihan, *Radiation Excursions at the ORNL Critical Experiments Laboratory.* ORNL-2452* (1958); B. R. Leonard, Jr., *A Study of the Radiation Burst in the Hanford Homogeneous Reactor.* HW-24327 (1952); H. C. Paxton, *Nucleonics* **16,** 80 (1958).
2. L. Geller and H. F. Henry, *Analysis of a Possible Radiation Burst.* K-1095 (1953).
3. F. S. Patton *et al.*, *Accidental Radiation Excursion at the* Y-12 *Plant.* Y-1234 (1958).
4. J. O. Bloemke and M. F. Todd, *Uranium*^{235} *Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time.* ORNL-2127 (1957).
5. J. W. Boyle, C. H. Hochanadel, T. J. Sworski, J. A. Ghormley and W. F. Kieffer, *Proceedings of the International Conference on Peaceful Uses of Atomic Energy Geneva,* 1955, Vol. VII, Paper 741. United Nations, New York (1956).

* This and similar designations refer to U.S. Atomic Energy Commission Research and Development Reports available from the Office of Technical Services, U.S. Department of Commerce, Washington 25, D.C.

Health Physics Pergamon Press 1959. Vol. 1, pp. 373–378. Printed in Northern Ireland

METABOLISM OF RADIOACTIVE ISOTOPES WHICH DEPOSIT MAINLY IN THE SKELETON*

BETSY J. STOVER
Radiobiology Laboratory,† University of Utah,
Salt Lake City, Utah

(*Received* 25 *September* 1958)

Abstract—Of the 102 known chemical elements more than half of them have been shown or can be predicted to deposit in the skeleton to a significant extent. These elements fall into three categories: the natural constituents of the skeleton, the elements which are chemically similar, and those which have no physiological analogs and which are generally characterized by low solubility at the physiological range of pH. Fortunately not all of these fifty to sixty elements are of importance in the problem of radiotoxicity, for nuclear properties eliminate some of them. The long-lived radioactive isotopes Ra^{226} and Pu^{239}, both of which decay by α-emission, have been extensively studied in beagle dogs. Ra^{226} is chemically similar to calcium, a major constituent of bone, while Pu^{239} has no physiological analog. The metabolism of the two isotopes following intravenous injection is compared from a chemical and a dosimetric point of view.

INTRODUCTION

Of the 102 known chemical elements more than half of them have been shown or can be predicted on a chemical basis to deposit in the skeleton to a significant extent. However, of these fifty to sixty elements which have chemical properties such that they go to the skeleton, not all have the requisite nuclear properties to become health physics problems. For example, element 101, mendelevium, is a member of a chemical group of elements which go to the skeleton, and thus it can be predicted to do likewise. But, since only a few atoms of this element are produced in each experiment,[1] the current role of mendelevium as a health hazard is rather small. The other extreme is illustrated by oxygen, which is the most abundant element both in the earth's crust and in the body. Yet it is not a health physics problem since its longest lived radioactive isotope, O^{15}, has a half-period for radioactive decay of only 2 min. Thus, both the chemical and nuclear properties of a radioactive isotope must be considered in evaluating its possible radiotoxicity.

GENERAL CLASSIFICATION

To compare the metabolism of these fifty to sixty elements which deposit in the skeleton let us classify them into three groups. For the first category we choose the light elements which are the normal constituents of the skeleton. These are hydrogen, carbon, nitrogen, oxygen, fluorine, sodium, magnesium, phosphorus, sulfur, chlorine and calcium. Some important radioactive isotopes of this group are H^3, C^{14}, Na^{22}, Na^{24}, P^{32}, S^{35}, Cl^{36}, and Ca^{45}. These isotopes are of great value in tracing the normal metabolic pathways and in studying metabolic kinetics; considerable knowledge has been gained through their use in biological and medical research. The mineral portion of bone is hydroxyapatite, which is a basic calcium phosphate, and accordingly it is the site of deposition of Ca^{45} and P^{32} (for a recent detailed account see Neuman and Neuman[2]) S^{35} is

* Presented at the Symposium on Internal Dose, Dr. K. Z. Morgan, Chairman, at the Third Annual Meeting of the Health Physics Society at Berkeley, California.

† AEC Contract AT(11-1)-119.

incorporated into cartilage and the ground substance of bone in muco-polysaccharides such as chondroitin sulfate. While H^3, C^{14}, Na^{22}, Na^{24} and Cl^{36} will be found in the skeleton, the important aspect of their metabolism in the evaluation of radiotoxicty is their ubiquity, for they are widely distributed in organic compounds and/or body fluids.

The second group of elements to be considered consists of the remaining alkaline earths, beryllium, strontium, barium and radium. The important feature of this group is their chemical and therefore metabolic similarity to calcium. Radioactive alkaline earth isotopes such as Sr^{90}, Ba^{140} and Ra^{226} deposit primarily in the mineral portion of the skeleton and the general aspects of their metabolism are analogous to those of calcium. There are differences, but they are of degree rather than of kind. In aqueous solution, all of the alkaline earths are divalent. There is a progressive increase in ionic size which is accompanied by a corresponding decrease in tendency to form complex ions. This undoubtedly is an important factor in the small metabolic differences observed among the alkaline earths.

The third category consists of those elements which have no physiological analog. Included are scandium, yttrium, lanthanum, actinium, the lanthanides, the actinides, a few of the transition elements (titanium, vanadium, zinc, zirconium, niobium and hafnium), gallium, tin and lead. In most cases the ions of these elements are smaller and have a higher charge than the alkaline earth ions. Consequently, the tendency to form complex ions is much greater, and many are insoluble at physiological pH's.

Calcium and calcium-like elements

Now that we have classified the elements which go to the skeleton according to their similarity and lack of similarity to calcium, what are the important general aspects of calcium metabolism? Many calcium salts are soluble at physiological pH's and there is a significant concentration of calcium in blood plasma. It occurs as calcium ion, and as complexes with the proteins and the organic acid anions of the blood. Transfer of calcium between blood and bone occurs with the rapidity characteristic of ionic reactions. Excretion of calcium is principally by the gut; for the dog, MALETSKOS and coworkers[3] observed a fecal to urinary excretion ratio of ten for endogenous calcium. CHEN and NEUMAN,[4] again studying the dog, found that 99 per cent of the calcium in the glomerular filtrate is reabsorbed by the tubules. The mineral portion of the skeleton is the principal site of deposition of calcium and calcium-like elements; retention by soft tissues is negligible in comparison. Calcium is incorporated into the skeleton by two processes. The first is characterized by a net transfer of calcium from the solution to solid phase. This occurs in growing bone and in the reconstruction phase of the continuous remodelling of bone. (In the resorption phase there will be a transfer of material from bone to the fluids.) The second process is ion exchange. Ions from the solution phase go to the solid phase, and an equal number move in the opposite direction so that there is no net transfer of material from fluid to bone. Metabolism of calcium is markedly influenced by age. The young growing animal is adding calcium to his skeleton, and even in the adult animal there is a difference in retention between the younger and older mature skeletons. Nutrition and general state of health are also important factors in calcium metabolism.

An exceedingly important member of this group is Ra^{226}, for a number of humans have incorporated radium into their skeletons and the effects have been studied. Many of the m.p.l.'s for various radioactive isotopes are derived in part from our knowledge of human radium poisoning.[5] The chemistry of radium is rather less complicated than that of many of the elements which deposit in the skeleton, for it exhibits but one valence state in aqueous solution, and, since it is the largest of the alkaline earth ions, it does not form complexes as readily as the others. However, Ra^{226} dosimetry is difficult because of its non-uniform deposition and the escape of a fraction of its radon daughter (Em^{222}).

Physiologically foreign elements

A general metabolic characteristic of those elements which are unlike calcium is deposition

on bone surfaces: endosteal, periosteal and trabecular. Often there is also significant deposition in soft tissues such as liver, kidney and spleen. These isotopes complex with plasma proteins and leave the blood more slowly than do the alkaline earths. Retention is usually high, and in general they are "stickier" than the alkaline earths. Age, nutrition and general health do not play as important a role in the skeletal metabolism of these isotopes, and there is more variation among members of this group.

Pu^{239} is an important example of the physiologically foreign elements. The rapid growth of plutonium production and processing facilities requires the best possible knowledge of Pu^{239} toxicity as a basis for the m.p.l. for Pu^{239}. Chemically, plutonium is unique in that four valence states can coexist in significant amounts in aqueous solution. Also, its small ionic size and high charge result in extensive hydrolysis and complex ion formation. When present in sufficient amount, plutonium is insoluble at pH 7. Pu^{239} decays by α-emission to U^{235}; thus, the dosimetry is not complicated by short lived daughters.

Comparison of Ra^{226} *and* Pu^{239}

At the Radiobiology Laboratory of the University of Utah, we are comparing the chronic toxicities of Ra^{226} and Pu^{239} in adult beagle dogs. It is anticipated that the results of these experiments can be used in conjunction with the known effects of Ra^{226} in humans to reevaluate the m.p.l. for Pu^{239} in humans. As part of this program we have made extensive studies of the metabolism of these two isotopes in dogs.[6,7]

Blood. The dogs are given the radioactive isotope in a single intravenous injection. The plutonium is prepared so that it is essentially all tetravalent, and the radium, of course, is divalent. In both cases the injection solution is a citric acid–sodium citrate buffer pH = 3.5 and 0.08 M in total citrate. During the first day following injection the plasma concentration of radium decreases much more rapidly than does that of plutonium. From plasma concentration data and average total plasma volume[8] of 450 ml for a 9 kg dog, we have calculated power functions for the amount circulating as a function of time during the first day. These are shown in Table 1.

From these equations, we note that at the time of our earliest measurement, 1 min after injection, more than half of the radium has left the blood compared with about 13 per cent of the plutonium. During the following interval from 1 min to 10 hr radium decreases by a factor of 18 while plutonium drops only by 2. In fact, plasma concentration of plutonium at 10 hr is roughly equal to that of radium at 1 min. This great difference in the rates at which the two radioactive isotopes leave the blood results from the difference in their chemical state in the blood. Dialysis studies[7,9] have shown that more than 90 per cent of radium in beagle blood is diffusible but less than 10 per cent of plutonium is diffusible. Thus injected radium is rapidly diluted in the body fluids in contrast with the protein bound plutonium.

Excretion. We have first directed our attention to the blood for two reasons: first, the radioactive isotopes we are studying are introduced directly into the blood from whence they go either to certain tissues or to the excretory systems, and, second, after the amount circulating becomes small compared with that in the tissues, the blood is the pathway from tissues to the excretory systems. Then the decrease in plasma concentration shows the same time dependence as does the decrease in excretion rate. In both cases fecal excretion is greater. For radium the fecal to urinary excretion value is 2.5; for plutonium during the first few weeks the value is 2.1 but it subsequently drops to 1.4. Radium is excreted much more efficiently, the daily excretion being thirteen times the amount circulating, whereas long term daily plutonium excretion is only two-thirds the amount in the blood. Renal clearance values are 1690 ml plasma/day for Ra^{226} and only 40 ml plasma/day for Pu^{239}.

Retention. Soft tissue deposition of Ra^{226} is transient, and, after a few days total retention and skeletal retention are essentially equal. This is not the case for Pu^{239}, for a significant fraction is deposited in the liver and persists for long times, excretion and retention of the two are compared in Table 1.

Table 1. *Metabolism of* Ra^{226} *and* Pu^{239} *in young adult beagle dogs**

	Ra^{226}	Pu^{239}	Units
(1) Amount in blood during first day	$41t^{-0.45}$	$87t^{-0.11}$	% dose, min
(2) Non-dialyzable fraction in blood	<0.1	>0.9	—
(3) Renal clearance	1690	40	g plasma/day
(4) Fecal to urinary excretion ratio	2.5	1.4	—
(5) Retention	$79t^{-0.20}$	$90 - 0.43t^{0.52}$	% dose, days
(6) Skeletal retention	$79t^{-0.20}$	$72t^{-0.04}$	% dose, days
(7) Location in bone	Mineral	Surfaces	—
(8) Liver retention	Negligible	$28t^{-0.07}$	% dose, days

* The results presented in this table are from references (6) and (7).

If we measure the variations in concentration of Ra^{226} and of Pu^{239} along the long axis of a long bone, we find that they are similar in that the maxima occur in the metaphyseal areas and the minima in the cortical bone. However, this similarity is misleading, since quite different modes of deposition lead to an apparently similar pattern on a macro-scale. Concentration of radium in bone is highest where remodelling of bone occurs, where the smallest crystals occur, and where the fluids have best access to the crystal surfaces. Plutonium deposits on bone surfaces and thus the higher concentrations occur where the surface area per unit weight is greater. Both modes of deposition lead to higher concentrations in trabecular than in cortical bone.

Dosimetry. From the retention equations of Table 1 and the knowledge that the fractional retention of radon (Em^{222}) (produced in the decay of Ra^{226}) is 0.20 in adult beagle dogs, dose rates are readily calculated.[6,7] The average skeletal dose rate* from Ra^{226} and its daughters resulting from the injection of 1 μc Ra^{226} into a dog which has a 900 g skeleton is

$$0.39t^{-0.20} \text{ rads/day} \qquad (1)$$

The average skeletal dose rate when 1 μc Pu^{239} is injected is

$$0.21t^{-0.04} \text{ rads/day} \qquad (2)$$

and the average dose rate to the liver is

$$0.25t^{-0.07} \text{ rads/day} \qquad (3)$$

(t is days after injection.) The change with time of the dose rates differs markedly. From equations (1) and (2) we see that the Ra^{226} and Pu^{239} skeletal dose rates per μc injected are equal at 46 days and thereafter that for Pu^{239} is greater. Integration of equations (1) and (2) shows the cumulative average skeletal doses per μc injected become equal at 4 to 5 months after injection, and at subsequent times that for Pu^{239} is greater.

The long term toxic effects of Ra^{226} and Pu^{239} are skeletal ones, ranging from subtle changes in bone architecture to spontaneous fractures and osteogenic sarcomas. Correlation of these effects with adsorbed radiation dose is a major aim of these experiments. The comparison of the effects of Ra^{226} and Pu^{239} will be facilitated by the fact that the beagles are all about the same age at injection, and throughout their entire lives their environment (including nutrition, and immunological and disease history) will be as nearly identical as possible. In view of the metabolic differences of Ra^{226} and Pu^{239} discussed above, it does not seem likely that equal skeletal effects of the two isotopes† will result from equal average skeletal doses. In comparing skeletal effects with skeletal doses the following three factors must be considered.

* Average dose is defined as that dose which would result if the energy were dissipated uniformly in the tissue.

† Preliminary results indicate this anticipated difference to be real.

(a) As a consequence of the different patterns of deposition in the bone, it appears that a larger fraction of the Ra^{226} energy is dissipated in the relatively inert bone mineral than is the case for Pu^{239}. (b) The change with time of average skeletal dose rate differs for the two isotopes. (c) In the case of Pu^{239}, the average dose rate to the liver is essentially equal to the average skeletal dose rate, while essentially all of the Ra^{226} dose is to the skeleton.

GENERAL COMPARISONS

Although a very great number of metabolic studies have been done with radioactive isotopes, the correlations which can be made are still rather limited. This results from the overwhelming number of factors involved, such as the large number of elements to be studied; the effects of chemical form, route of administration, and duration of exposure; the differences among species, and the effects of age, general health, etc., within a given species.

Since the calcium-like elements are more similar chemically and have a physiological analog, greater metabolic similarity is anticipated than for the physiologically foreign group. Studies of Sr^{90} in adult beagles[10] and Ca^{45} in adult mongrel dogs[3] substantiate this. Retention is essentially all skeletal and is very similar for Ca^{45}, Sr^{90} and Ra^{226}. There are differences in plasma protein binding and renal clearance which appear to correlate with ionic radii. Retention of all three alkaline earths by the dog is greater than human* radium retention.[11]

The elements included in the physiologically foreign group differ much more chemically and thus cannot be as well represented by a single isotope. The solution chemistry of thorium is quite similar to that of tetravalent plutonium and we have studied Th^{228} (RdTh) metabolism in adult beagle dogs.[12] The skeletal retentions of Th^{228} and Pu^{239} are quite similar, but the soft tissue deposition differs. A larger fraction of Pu^{239} deposits in the liver, but the kidney Th^{228} deposition is greater.

An interspecies comparison of Pu^{239} shows that retention by the mouse is less than that by the dog which is less than human retention.[7,13,14] The initial deposition pattern is quite similar for all three, but in the mouse the liver Pu^{239} decreases quite rapidly in contrast with its persistence in dog and man.

In conclusion, there appears to be a need for more comparative studies in which a chemically related series of isotopes is studied in one species under similar conditions, and in which a given isotope is studied in several different species. Experiments of this kind should yield correlations which would enable us to make more valid extrapolations to elements or species which have not been studied.

SUMMARY

(1) Although over half the known elements are potential "bone-seekers" because of their chemical properties, the actual number which are of importance in the problem of radiotoxicity is limited by nuclear properties.

(2) The radioactive isotopes which deposit mainly in the skeleton fall into three categories: natural constituents, alkaline earths which are chemically like calcium, and those which have no physiological analog.

(3) Examples of the second and third categories are Ra^{226} and Pu^{239}, respectively. The metabolism of these two isotopes in young adult beagle dogs has been compared from a chemical and a dosimetric point of view.

REFERENCES

1. A. Ghiorso, B G. Harvey, G. R. Choppin, S. G. Thompson and G. T. Seaborg, *Phys. Rev.* **98,** 1518 (1955).
2. W. F. Neuman and M. W. Neuman, *The Chemical Dynamics of Bone Mineral.* The University of Chicago Press (1958).
3. C. J. Maletskos, *Progress Report, AEC Contract* AT(30–1)–952, pp. 19–40. The Radioactivity Center, Massachusetts Institute of Technology (1954).
4. P. S. Chen and W. F. Neuman, *Amer. J. Physiol.* **180,** 623 (1955).
5. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water.* USNBS Handbook 52 (1953).
6. M. A. Van Dilla, B. J. Stover, R. L. Floyd, D. R. Atherton and D. H. Taysum, *Rad. Res.* **8,** 417 (1958).

* Note that the humans received multiple injections; the dogs single injections.

7. B. J. Stover, D. R. Atherton and N. Keller, *Rad. Res.* In press.
8. J. E. Parkinson and J. H. Dougherty, *Proc. Soc. Exp. Biol. Med.* **27,** 722 (1958).
9. B. J. Stover and N. Keller, *University of Utah Radiobiology Laboratory Progress Report* (AECU–3583), pp. 77–81 (1957).
10. B. J. Stover and D. R. Atherton, *Proc. Soc. Exp. Biol. Med.* **99,** 201 (1958).
11. W. P. Norris, T. W. Speckman and P. F. Gustafson. *Amer. J. Roentg.* **73,** 785 (1955).
12. B. J. Stover. Unpublished results.
13. W. H. Langham, *Amer. Ind. Hygiene Assoc. Quarterly* **17,** 305 (1956).
14. W. H. Langham, S. H. Bassett, P. S. Harris and R. E. Carter, *Distribution and Excretion of Plutonium Administered Intravenously to Man.* LA–1151 (1950).

Health Physics Pergamon Press 1959. Vol. 1, pp. 379–389. Printed in Northern Ireland

EXTERNAL β-DOSES FROM RADIOACTIVE FALLOUT

G. M. DUNNING
Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington 25, D.C.

(*Received* 10 *July* 1958)

Abstract—The purpose of this article is to present information on external β-radiation doses from fallout material and to make some evaluation of their biological hazard. A brief review is given of past laboratory data on external radiation exposures and their effects upon the skin. Consideration is given to estimating potential external β-doses from fallout based on field data and theoretical calculations. Instrument evaluation for β-monitoring is touched upon briefly. Lastly, some suggestions are proposed as guides in developing radiological safety criteria for external β-irradiation from fallout material.

I. OBSERVED SKIN DAMAGE FROM FALLOUT

GAMMA-rays constitute the external radiation of principal concern associated with radioactive fallout. However, β-radiation doses may also be significant, especially when the fallout material remains in contact with the skin. So-called "β-burns" occurred on a few cattle in 1945 and 1952, on horses in 1953, and one horse in 1955, all of which were grazing within 20 miles of test nuclear detonations in the United States. β-burns occurred also on some of the inhabitants of the Marshall Islands (Table 1)[1], on fishermen aboard the Japanese vessel, *Fukuryu Maru*, and on some U.S. Test personnel following the 1 March, 1954 detonation in the Pacific. In all cases, the detonations were surface or near-surface bursts resulting in the radioactive isotopes formed becoming associated with dirt particles. These were first swept high into the air at the time of the explosion and then the larger particles fell to the earth in the nearer areas. Some of these were deposited on the skin of humans and animals.

One of the principal facts that has been learned from past experience with fallout is that recognizable β-skin damage has occurred principally on the areas of the body where the radioactive material actually remained in contact with the skin. There was no evidence of skin damage where even a single layer of cotton clothing was present between the radioactive material and the skin. It is possible that erythema could have occurred but the dark skin of the Marshallese may have masked this effect. However, more serious skin damage would have been observed, if present. Since it would be expected that absorption of the betas from fission products through such a thickness of clothing would attenuate them by only about 20–25 per cent, it would appear that these minimal β-doses probably resulted from three principal factors: (a) the greater loss of the material from cloth surfaces than from the moist skin, (b) the holding of the material away from the skin, and (c) the presence of soft betas from induced activities. The β-burns on the Marshallese were definitely of more superficial nature than those on the animals in Nevada.

Table 1. *Exposure data on Marshallese*

Number of personnel involved	External γ-dose (r)	Deep lesions	Superficial lesions	No lesions	Epilation
64	175	13	45	6	35
18	69	1	15	2	3
157	14	0	0	0	0

As would be expected, the greatest skin damage on the Marshallese occurred when the fallout material remained in contact with the skin. What may be more difficult to understand is the absence of any marked effect solely from the β-emissions from the material on the ground except that some lesions were observed on the knees of children. However, these probably were the result of kneeling on contaminated surfaces. SONDHAUS[1] calculated that if there had been no shielding the most highly exposed Marshallese (175 r whole body-γ) would have received an *additional* 2000 reps (β) at the level of the dorsum of the foot, 600 reps at hip height and 300 reps at the head. The higher values should have been adequate to produce observable effects. The resolution of this uncertainty probably does not lie as much in re-evaluation of biological effects versus actual doses received, since these are fairly well documented from laboratory experiments, as in further study of the physical parameters involved including more precise information on the beta energy spectra and the nature of the exposures.

Additional cases of human β-burns from fallout occurred on some Japanese fishermen aboard the *Fukuryu Maru*.[2] These lesions were similar to those of the Marshallese. There also were some American service personnel who received minor skin injury from the same incident. One additional human case of β-burn from fallout material was an Air Force officer in charge of transportation of radioactive samples from the Pacific Proving Ground to the United States;[3] a lesion developed on the forehead and right eyebrow region. The skin damage showed a normal repair history but the previously black color of the eyebrow was replaced by white hair upon regrowth.

It is not clear as to the exact degree of contamination of the areas in which the cattle and horses were grazing following the test detonations of 1945, 1952, 1953 and 1955. They were all within 20 miles of ground zero (tower bursts) and this, together with monitoring data at the Nevada Test Site, indicated that the concomitant external γ-exposures probably were over 75 r. This falls into the range of crude estimates made above (Table 1) for the Marshallese.

II. RADIATION DOSES AND SKIN DAMAGE

Table 2 is a condensation of a few representative experimental studies made concerning skin reactions to external radiation sources. It is difficult to place them on a common basis for comparison due to differences in:

(1) Depth dose curves for different energies and types of radiation.
(2) Type of radiation used: (a) photon (X-radiation) or (b) corpuscular (β-radiation).
(3) The type of animal, the body area selected for irradiation, and the criteria used (microscopic recognizable epidermal atrophy, tanning, mild erythema, dry desquamation, epilation, ulceration, etc.).

For example, the dose required to produce a given skin effect varies with the wavelength of X-rays used,[10] and in addition is a function of the kind of radiation employed.[12] The differences between surface doses of beta emitters that produce transepidermal injury are nicely illustrated by Table 3.[13] As the authors pointed out, whereas the surface doses vary considerably, the dose to produce microscopically recognizable transepidermal injury at a depth of 0.09 mm (thickness of epidermal layer of porcine skin) is approximately constant at 1400 ± 300 reps.

The depth–dose relationship may also account partially for differences in doses between X-rays and β-rays required to produce epilation. The depth of the hair follicle varies, but as a first approximation, if a depth of 1 mm is considered, the dose at this depth from a 200 kV P X-ray is essentially the same as at the surface, while the β-dose at this depth from P^{32} may be only 40 per cent of its surface dose.

The possibility that X-radiation and beta radiation may require different doses to produce the same biological effect is suggested in Table 2.[4] Using 30 kV X-rays and P^{32} (on rabbit ears) it was noted that "· · · it required approximately three times the dose of β-rays as compared with low-voltage X-rays to produce a minimal erythema, about four times the dose to cause the first signs of epilation, and about four and a half times the dose to produce epilation in all the animals treated". Whereas, these were surface doses, it may not be that all of this difference is

Table 2. Skin reaction to external radiations

Reference No.	Subject exposed	Source of radiation	Dose rate of application	Total doses (reps)	Scaliness			Epilation		Erythema		
					Appearance time (days)	Duration (days)	Showing reaction (%)	Appearance time (days)	Showing reaction* (%)	Appearance time (days)	Duration (days)	Showing reaction (%)
(4)	Rabbit (ears)	30 kV		1400	28	36	66	32	100	28	26	90
				900	0	0	0	41	33	0	0	0
				800	0	0	0	0	0	0	0	0
		P^{32} (1.7 MeV)	145 r min	3500	20	42	50	48	25†	18	12	50
				3000	25	13	60	0	0	18	15	40
				1500	0	0	0	0	0	0	0	0
					Effects							
(5)	Porcine skin	Sr^{90}–Y^{90} Plaque 0.61 MeV 2.2 MeV	9.3 reps/sec	(A) 1500–2000	(A) Microscopically recognizable epidermal atrophy. Time of appearance: 1–2 weeks. Duration: 2–4 weeks							
				(B) 2000–2500	(B) Epidermis exfoliated (erythema frequently preceded and epilation usually accompanied by permanent destruction of hair follicles and sebaceous glands). Time of appearance: during second week. Duration: Several weeks. Majority showed continued chronic radiation dermatitis for as much as 3 months and longer.							
				(C) 4000 and higher	(C) Ulceration. Time of appearance: Varied greatly from days to weeks, may be masked by exfoliation with crust formation.							
(6)	Sheep shaved skin on side of animal	Sr^{90}–Y^{90}	33 reps/sec	2500	Superficial lesions developed in one sheep but not a second in 13 days. No appreciable cessation of wool growth in second. Exposed areas not recognizably different after 19 weeks.							
				5000	Full thickness of epithelium not destroyed.							
				15,000	Loss of superficial epidermis in outer one-third to one-half of the wool fiber follicles followed by regeneration. Areas still bald after 1 year.							
				25,000	Full thickness of dermis and whole follicle lost. First seen in 8 days. Healing complete in 8–9 weeks.							
	Sheep with applicator placed lightly over wool	Sr^{90}–Y^{90}	5400 reps/min	5400								
				16,200	In 10 days it was possible to pluck out the wool from areas receiving 16,200 reps and higher.							
				48,000								
				145,000	An ulcer developed in one of two sheep at the area receiving 145,000 reps.							
(7)	Humans	200 kV P X-rays	—	400–500	Temporary epilation in about 3 weeks.							
				700 or more	Permanent epilation.							

Table 2. (continued)

Reference No.	Subject exposed	Source of radiation	Dose rate of application	Total doses (reps)	Effects
(8)	Humans volar aspect of forearm and inner aspect of mid-thigh	P^{32} 1.7 MeV	2.5 to 5 min exposure	140 170 200 250 635 813 1000 1180	Tanning of skin. Mild tanning in 60% of those exposed to 170 reps and 80% for 200 reps with some cases starting within 48 hr. Persisted in some for as long as 10 weeks. Erythema. Threshold (mild) erythema in 60% of those exposed to 635 reps and 80% for 813 reps. Appeared within first day for all doses of 813 through 1180 reps and persisted for one day for 813 reps dose and 2 to 4 days for 1180 reps dose. Secondary reaction: Definite erythema by twenty-first day in less than 20% of cases for 635 reps and 90% for 813 through 1180 reps. Persisted: 60 days for 813 and 1000 reps, 70 to 80 days for 1180 reps. No cases showed bleb formation, but at 1180 reps minute vesicles formed in 30% of cases on fifth and sixth week followed by dry spotty desquamation of the most superficial layers of epithelium.
(9)	Humans: volar aspect of forearm	P^{32} (1.7 MeV)	~6‡ rep/hr	143‡ (24 hr exposure)	Threshold erythema dose discernible in 5 days. No noticeable difference between irradiated and unirradiated skin in 68 days.
			~75‡ rep/hr	7200‡ (96 hr exposure)	Erythema starting on third or fourth day after beginning of application, 18–20 days afterward dry scalp epidermite occurred which subsided in 35–45 days.
			~235 reps/hr	17,000 (72 hr exposure)	Initial reaction same as 7200 reps but a bullous epidermolysis resulted in 16–20 days after beginning of exposure; 50–60 days from beginning of exposure, the skin recovered; after 6 months slight dimpling of skin noted in area exposed.
(10)	Mice	P^{32} (1.7 MeV)	Single exposure	<3000	No increase in frequency of tumors.
(10)	Mice	P^{32} (1.7 MeV)	Single exposure	3000 4500 rep	Tumor induction significantly greater than spontaneous.
	Rats	P^{32}	Single exposure	4000 rep and higher	Conspicuous increase in tumor production.
(11)	Rats	P^{32}	50 reps/day for 9 months	~18,500 rep	Occasional ulceration (exclusively in subcutaneous region).
	Rats	P^{32}	5 reps/day for 9 months	~1850	No discernible effects.

* 1000–1400 r majority permanently epilated; 1400–2000 r all permanently epilated.

† 600 r and above permanently epilated.

‡ Computed as an average to depth of first millimeter of tissue.

Table 3. Amount of transepidermal radiation required for the production of recognizable transepidermal injury (porcine skin)

Isotope	Energy (MeV)	Surface dose required (rep)	Estimated dose at depth 0.09 mm (rep)
Sulfur-35	0.17	20,000	1200
Cobalt-60	0.31	4000	1600
Cesium-137	0.55	2000	1700
Yttrium-91	1.53	1500	1200
Strontium-90 Yttrium-90	0.61 2.20	1500	1400

accounted for on the basis of the greater attenuation of the betas. For example, it would be expected that the 1.7 MeV betas from P^{32} might be attenuated to about 90 per cent of the surface dose rate at 0.1 mm depth and to about 40 per cent at 1.0 mm depth. (The authors reported that approximately 25 per cent of the 30 kV radiation was absorbed in the rabbit ear, the thickness of which varied from 0.75 to 1.37 mm.) A later study suggests a relative biological effectiveness for P^{32} to be slightly less than one-half that for 200 kV X-rays for production of erythema in a rabbit's ear.[14]

The differences in biological response for various animals are illustrated by Tables 2 and 4.[7] Other evidence of this difference in response is reported by RAPER *et al.*; "The breakdown of apparently healed skin, forming extensive and severe ulcers, was of common occurrence after (single) doses (P^{32}) of 300 reps or more in mice and 7500 reps or more in rabbits."[15] Despite these differences, the other data suggest that valid comparisons may be made between man and some animals for certain effects. RAPER[16] stated, "Doses of 1000 and 1180 reps to human skin produced erythema and pigmentation comparable to the reaction in rabbits to 1000 reps." (β-doses).

Table 4. Doses producing permanent epilation

Animal	Dose (r)	Type of radiation and filtration	Reference
Human	700	—	(7)
Rabbit	2500	200 kV, 2 mm Cu+1 mm Al	(11)
Rabbit	2000	100 kV, 2 mm Cu	(11)
Dog	2400	120 kV, 1 mm Al	(11)
Mouse	1900	Hard	(11)

Since we are dealing with particulate matter when discussing radioactive fallout, one may raise the question of possible radiation effects from a single particle. Different methods of estimating β-dose rates as a function of distance from a point source have been proposed, but using any of these methods and the observed data on individual particle activity in fallout leads to the conclusion that it is possible for the more highly active particles to deliver sufficient doses over a large enough area (and volume) to produce recognizable skin damage. Several studies have been made of this "particular problem" with the one most pertinent to this discussion having been conducted by PASSONNEAU *et al.*[17] An area of 35 cm^2 of rats' skins were exposed for 48 hr. In one series the material was distributed uniformly over the source while in others the activity was in the form of point sources. They reported, "Rats exposed to the distributed sources showed at least a fivefold increase in development rate of tumors when the incidence was corrected for age. Malignant and benign tumor formation was greater after exposure to diffusely distributed β-radiations than after exposure to the same amount of isotope in point sources, suggesting that exposure to 'hot' particulate matter does not entail an unusual hazard due to spatial concentration of radioactivity."

Another factor for consideration in estimating beta doses from fallout material is the fact that the doses are not delivered over a period of time. Since the skin is a relatively rapidly reparable tissue it might be expected that this could be a significant factor in relation to radiation doses from fallout. RAPER[11] reported; "At the end of 9 months of irradiation with 50 reps/day, mice showed slight epilation and occasional ulceration · · ·" and "Lower daily doses of β-rays, 5.0 and 0.5 rep/day to rats and 5.0 reps/day to mice, produced no discernible effects."

The effects of fractionation of doses were investigated by many including MacComb and Quimby. Using data developed by Duffy *et al.*[18], they calculated that recovery from skin doses from 200 kV X-ray set in promptly, and after 6 hours, 36 per cent of the effect had been lost, and after 12 hours, 50 per cent, followed by decreasing recovery rates.[19] After the second day the recovery proceeded at a rate of 25 per cent daily up to the fifth day (the end of the data). A similar analysis of data from Reisner[20] also indicated a 25 per cent per day recovery rate up to 7 days. For daily treatments up to 12 days, the residual effect progressed slowly up to about 85 per cent and for 27 days, up to 95 per cent.

In 1944, Strandgvist[21] found that 100 to 175 kV P X-rays given at 30 to 200 r/min in a single treatment (D_0) were equivalent in their effect on normal or malignant human skin to a total dose of 1.26 D_0 given in two exposures 1 day apart. When plotted on log–log paper the total dose vs. days elapsed between initial and final treatment approximated straight lines out to 40 days (end of the data given). The equation for these curves is:

$$D = Kt^{0.22}$$

where: D = total dose (r)
K = dose in one treatment times 1.26
t = days elapsed between initial and final treatment

the values for K being about:

1230 for erythema
2000 for epidermatitis sicca
2820 for cure of skin carcinoma
3750 for skin necrosis

Strandgvist's treatment of the data does not provide for any irreparable damage factor but it would seem likely this would exist.

The similarity of biological effects from X-rays and β-rays has been noted above with the weight of evidence indicating that β-irradiation is less biologically effective on a rad dose basis. The relative skin rate of repair from damage produced by X-rays and β-rays in humans is not known, but the known differences in depth-dose curves would suggest at least equivalence, if not a faster rate of repair from β-irradiation than X-rays for equal surface doses.

Whereas, the effect of local β-doses to the skin has been fairly well documented, effects such as lethality from β-doses over the entire body is quite uncertain, at least for man. Reference 22 states "· · · the median lethal dose (whole-body, single exposure, P^{32}, in 45 days) · · · was considerably different for the three species: mice 4700 reps; rats 7500 reps, and rabbits about 17,500 reps." The median lethal dose (β) was thought "· · · to be proportional to the weight (volume) of the animal." The above-mentioned experiments provided relatively even distribution of the β-doses over the body, whereas those for an erect man in a fallout field might easily vary by a factor of 4 or more between the lower and upper extremities of the body. For a prone man, such as a soldier in a contaminated field, the β-exposure would be more uniform over the lower surface of the body.

III. β-DOSES FROM FALLOUT

The total β-radiation dose delivered is principally a function of the amount and specific activity of the material, the length of time it is in contact with the skin, the energy of the β-particles, and of the exposure geometry.

The β-activity of mixed fission products decays approximately according to the following relationship:

$$A = A_1 t^{-1.2}$$

where: A = β-activity at any time after detonation later than 1 unit.
A_1 = β-activity at any one unit of time after detonation
t = time after detonation (in same units as above)

By substituting dose rate and then integrating the above equation, accumulated total doses can be estimated. This approach does not take into account the relatively large change in energy of the β-particles from fission products between the first hour and fourth day as indicated in Fig. 1.[23] Therefore, Fig. 2 was constructed on the basis of changes in relative β-energies with time and using the decay scheme of Bolles and Ballou for mixed fission products.[21] Fig. 3 shows the accumulation of β-energies for varying times of initial contamination, based on these assumptions.

Theoretical treatment has been given to

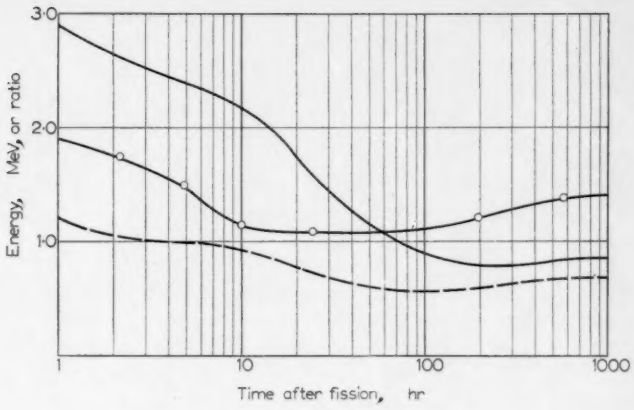


FIG. 1. Calculated average maximum values. ——— β; – – – γ; -o-o- ratio-β-particles per sec/γ-photons per sec.

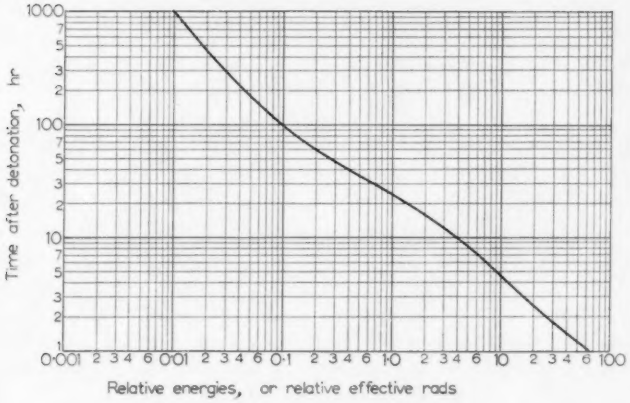

1000
100
10
1
Time after detonation, hr
0·001 2 3 4 6 0·01 2 3 4 6 0·1 2 3 4 6 1·0 2 3 4 6 10 2 3 4 6 100
Relative energies, or relative effective rads

FIG. 2. Relative decay of β-energies (normalized to 1 day).

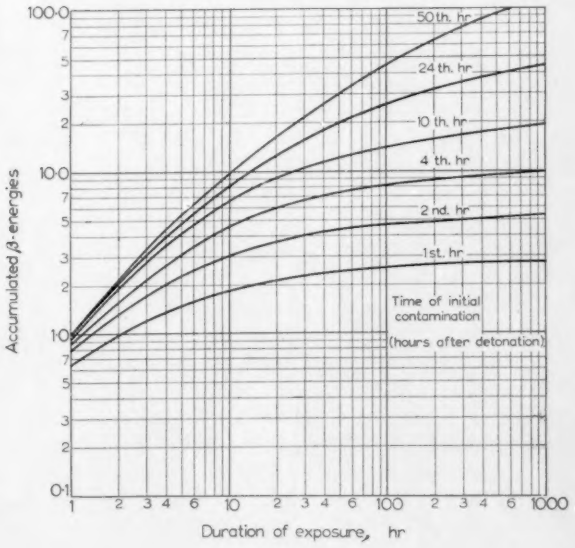


FIG. 3. Accumulation of relative β-energies for varying times of initial contamination. (Based on the unit/hr at time of contamination.)

absorption coefficients,[25,26] and considerable experimental work has been performed for single β-emitting isotopes. Some absorption curves for different fallouts at varying times are given in reference 27. As would be expected, β-depth dose curves for mixed fission products do not coincide with those of a single isotope. Also, varying amounts of induced activities would change the β-energy spectra.

These relatively large variances can make significant differences in doses delivered to the basal layer of the epidermis and to greater depths. For example, it was estimated that 50–80 per cent of the β-activity from the fallout on the Marshallese was in the range of 100 KeV and 20–50 per cent around 600 KeV.[1] Whereas the lower energy betas were attenuated much more in passing through the epidermis, yet their abundance and higher specific ionization at the basal layer probably was the major cause of the superficial skin lesion. Also, it may be argued that estimating doses to the basal layer of the epidermis does not fully evaluate the biological hazard, since the more energetic betas will irradiate deeper tissues increasing the risks of epilation, deeper necrosis and of skin cancer.

Only as a first approximation one may compare doses from P^{32} with mixed fission products. There are data[28,29] to suggest that 1.0 $\mu c/cm^2$ of P^{32} has an estimated surface β-dose rate of 5.9 reps/hr and 5.05 reps/hr, respectively, with an average value of about 5.5 reps/hr (5.1 rads/hr). Based on an assumed β to γ-ratio of 1 : 1, this degree of contamination (1 $\mu c/cm^2$) spread uniformly over an infinite field would produce approximately 0.1 r/hr of γ-radiation at a height of 3 ft above the ground.

In addition to direct contamination of the body, is the problem of β-radiation emanating from fallout material remaining on the ground. Much consideration has been given this problem of "β-bath" since such large areas of the body might be involved. A rough analogy of the potential hazard is when significant areas of the body surface suffer thermal burns.

Field measurements, using miniature ionization chambers of 8 mg/cm^2 wall thickness in phantoms, indicated that the relative ionizations caused by β and by γ-radiations at a height of 5 cm was about 20 to 1 on the day of the detonation with this ratio dropping to about 10 to 1, and then 5 to 1 on the first and second days, respectively, afterwards.[30] As the height above ground increased these ratios likewise decreased, showing a factor of 4–6 difference between 5 to 180 cm heights. Theoretical calculations[1,31] fall roughly in this range of values.

IV. INSTRUMENT EVALUATION

Assuming a given uniform distribution of activity, there are three major factors that influence a beta survey meter reading:

(1) Area of contamination.
(2) Distance between the contaminated surface and the instrument.
(3) Energy spectra of the betas.

The contamination from fallout is not uniform. A survey instrument will in a sense integrate the radiation it sees and record a certain value, yet a non-uniformly contaminated surface may vary significantly in the degree of local (small area) exposure rate. A small advantage may be gained by moving the survey meter over a given contaminated area to note variations, and/or small probes may be utilized to better estimate the variance in the degree of contamination.

To illustrate the possible use of a field instrument to measure β-radiation, three CD V-720 instruments were first calibrated with radium (γ) and then used to obtain ionization currents as a function of known β-activity per unit area deposited on plaques. Then using an extrapolation chamber, doses at a depth of 7 mg/cm^2 were estimated. The data are summarized in Table 5.[32] CD V-720 has a 40.5 mg/cm^2 β-window which will greatly attenuate the soft betas. A window of 7 mg/cm^2 will more closely approximate the thickness of human epidermis for the major portion of the body.

It may be inferred from these data that for a given area of contamination a change in the distance between the instrument and the contaminated surface can cause a significant change in r/rad ratio, with a greater relative change occurring when the area is smaller. Also, with a constant instrument-surface distance, a change in area of contamination can also cause a significant change in the r/rad ratio. The data in Table 5 also suggest that the shape of the contaminated surface, except for extreme

Table 5. Beta-ray calibration. FCDA CDV-720 survey meters†* (r/rad)

Serial No. 2 Source size (in.)	Distance from source to meter (in.) 6	4	2	1
	Phosphorus-32 (1.7 MeV)			
12 × 12	0.0412	0.0556	0.0739	—
6 × 12	0.0255	0.0375	0.0591	—
3 × 12	0.0137	0.0214	0.0384	—
1 × 12	0.0048	0.0080	0.0150	—
6 × 6	0.0157	0.0263	0.0484	—
3 × 3	0.0049	0.0091	0.0210	—
1 in. Dia.	0.0004	0.0009	0.0022	—
	Thallium-204 (0.762 MeV)			
12 × 12	0.012	0.019	0.025	0.031
6 × 12	0.008	0.014	0.019	0.027
3 × 12	0.004	0.008	0.012	0.017
1 × 12	0.001	0.002	0.004	0.006
6 × 6	0.006	0.010	0.017	0.023
3 × 3	0.002	0.004	0.010	0.012
Serial No. 3				
	Phosphorus-32			
12 × 12	0.0404	0.0570	0.0761	0.0868
6 × 12	0.0260	0.0393	0.0603	0.0773
3 × 12	0.0135	0.0213	0.0403	0.0571
1 × 12	0.0049	0.0082	0.0147	0.0234
6 × 6	0.0159	0.0267	0.0493	0.0671
3 × 3	0.0049	0.0092	0.0218	0.0365
1 in. Dia.	0.0004	0.0008	0.0019	0.0032
	Thallium-204			
12 × 12	0.014	0.016	0.027	0.033
6 × 12	0.008	0.014	0.023	0.029
3 × 12	0.004	0.008	0.014	0.018
1 × 12	0.002	0.002	0.006	0.008
6 × 6	0.006	0.010	0.018	0.025
3 × 3	0.002	0.002	0.008	0.014
Serial No. 5				
	Phosphorus-32			
12 × 12	0.0396	0.0543	0.0726	0.0812
6 × 12	0.0239	0.0379	0.0593	0.0727
3 × 12	0.0135	0.0213	0.0390	0.0518
1 × 12	0.0048	0.0077	0.0136	0.0214
6 × 6	0.0118	0.0261	0.0490	0.0656
3 × 3	0.0047	0.0091	0.0214	0.0370
1 in. Dia.	0.0001	0.0008	0.0021	0.0031
	Thallium-204			
12 × 12	0.012	0.019	0.027	0.031
6 × 12	0.008	0.014	0.021	0.027
3 × 12	0.004	0.008	0.014	0.017
1 × 12	0.002	0.003	0.004	0.006
6 × 6	0.006	0.010	0.012	0.025
3 × 3	0.002	0.004	0.008	0.014

* Reference (31). † β-window thickness = 40.5 mg/cm^2.

examples, does not influence the r/rad ratio as greatly as does changes in total area. The presence of γ-radiation from fallout material on parts of the body probably would not affect significantly the meter reading while performing personnel beta monitoring in a clean area. For example, with a 6 in. × 6 in. area of contamination and the meter held at 4 in. from the surface, the γ-radiation for 1 day old fission products probably would contribute less than 10 per cent to the meter reading. Personnel monitoring in a contaminated field would require shielding of the instrument from the betas originating from other than the contaminated surface under consideration and possibly shielding from the associated gamma field if the latter is relatively high.

V. PRACTICAL APPLICATIONS

Estimates of the β-hazard from fallout have been made by others.[33–35] It would be desirable if all the before mentioned parameters and variables could be synthesized into one set of simple radiological safety criteria for the general population. The author has been unable to do so at this time, to an acceptable degree. Rather, the following are suggested as guides.

(1) The external γ-radiation is the dominant hazard over external β-radiation in a fallout field.

(2) In a fallout field, time-consuming personnel monitoring should be delayed in favor of other acts to reduce the γ-exposure (evacuation or use of special shelters).

(3) Fallout material remaining in direct contact with the skin is most likely to produce β-burns.

(4) Since a significant percentage of the total possible skin dose is delivered in the early times after detonation, it would be advisable to initiate personnel decontamination procedures as soon as possible. This would include not only the more elaborate procedures such as showering but also common-sense actions as brushing off the clothing and gently wiping off the exposed skin.

(5) When monitoring large numbers of people with limited decontamination facilities a gross evaluation probably will have to be made first, and priorities established to provide for

those most seriously contaminated. Later, decontamination may be continued to reduce the levels to minimal values for all.

(6) If entry must be made into a contaminated area, as much of the body as possible should be covered to prevent direct contamination of the skin and to provide a relatively quick and effective method of later removing the contaminant.

Acknowledgments—The author is grateful to Dr. W. Gross, Department of Radiology, Columbia University, for generous assistance in developing the data contained in the section "Instrument Evaluation," and to the following for their review of the manuscript and their many helpful suggestions: Dr. C. L. Dunham, Director, Division of Biology and Medicine, AEC, Washington, D.C.; Dr. F. Western, Assistant Director, Division of Biology and Medicine, AEC, Washington, D.C.; Dr. D. Bruner, Chief, Medical Branch, Division of Biology and Medicine, AEC, Washington, D.C.; Dr. S. Warren, Cancer Research Institute, New England Deaconess Hospital, Boston, Massachusetts; Dr. J. C. Bugher, Director, Medical Education and Public Health, The Rockefeller Foundation, New York,; Dr. K. Z. Morgan, Director, Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee; Dr. P. C. Tompkins, Scientific Director, and Drs. J. D. Teresi, G. A. Work and E. Tochilin, U.S. Naval Radiological Defense Laboratory, San Francisco, California; Dr. H. L. Andrews, National Institutes of Health, Bethesda, Maryland; Dr. A. M. Brues, Director, Division of Biological and Medical Research, Argonne National Laboratory, Lamont, Illinois; Dr. J. W. Healy, Hanford Operations Office, AEC, Richland, Washington; Dr. E. G. Williams, Bureau of Medical Services, Department of Health, Education and Welfare, Washington, D.C.; Dr. R. F. Brown, School of Medicine, University of California Medical Center, San Francisco, California; Dr. A. H. Dowdy, School of Medicine, University of California Medical Center, Los Angeles, California; Dr. L. H. Garland, St. Joseph's Hospital, San Francisco, California.

REFERENCES

1. E. P. Cronkite, V. P. Bond and C. L. Dunham (Editors), *Some Effects of Ionizing Radiation on Human Beings*. U.S. Government Printing Office, Washington, D.C. (1956).
2. Kentaro Shimizu, Koichi Ishikawa, Yagushi Saito, Koichi Nakamura, Toho Sato, Shiuchi Torada, Tadashi Sugiyama, Shigenobu Takayama, Hiroshi Huruya, Mosuke Onu and Hideo Inagaki, (Committee for Compilation of Report on Research in the Effects of Radioactivity), *Research in the Effects and Influences of the Nuclear Bomb Test Explosions* Vol. II, p. 1333. Japan Society for the Promotion of Science, Tokyo (1956).
3. R. A. Conard and C. F. Tessmer, *Amer. Med. Ass. Arch. Dermatol.* **74,** 663 (1957).
4. F. W. Bishop in *Biological Effects of External Radiation* (Edited by H. A. Blair) p. 194. McGraw-Hill, New York (1954).
5. A. R. Moritz and F. W. Henriques, *Lab. Invest.* **1,** No. 2, 167 (1952).
6. C. E. Lushbaugh, J. F. Spalding and D. B. Hale, *Comparative Study of Experimentally Produced Beta Lesions and Skin Lesions in Utah Range Sheep*. Los Alamos Scientific Laboratory, Los Alamos, New Mexico (1953).
7. S. Warren, *Arch. Pathol.* **35,** No. 2,340 (1943).
8. J. E. Wirth and J. R. Raper in *Biological Effects of External Beta Radiation* (Edited by R. E. Zirkle) p. 193. McGraw-Hill, New York (1951).
9. B. V. A. Low-Beer, *Radiology* **47,** No. 3, 213 (1946).
10. J. C. Hudson, *Radiology* **29,** No. 1, 95 (1937).
11. J. R. Raper, P. S. Henshaw and R. S. Snider in *Biological Effects of External Beta Radiation* (Edited by R. E. Zirkle) p. 212. McGraw-Hill, New York (1951).
12. C. W. Wilson, *Radiology* **57,** No. 1 (1951).
13. A. R. Moritz and F. W. Henriques, *Lab. Invest.* **2,** No. 2, 167 (1952).
14. W. B. Seaman, M. C. Terpogossian and W. B. Itiner, *Radiology* **65,** No. 2, 260 (1955).
15. J. R. Raper, P. S. Henshaw and R. S. Snider in *Biological Effects in External Beta Radiation* (Edited by R. E. Zirkle) p. 200. McGraw-Hill, New York (1951).
16. J. R. Raper, J. E. Wirth and K. K. Barnes in *Biological Effects in External Beta Radiation* (Edited by R. E. Zirkle). McGraw-Hill, New York (1951).
17. J. V. Passonneau, A. M. Brues and J. A. Hamilton, *Proc. Amer. Ass. Cancer Res.* **1,** No. 1 (1954).
18. J. J. Duffy, A. N. Arneson and E. L. Voke, *Radiology* **23,** No. 4, 486 (1934).
19. W. S. MacComb and E. H. Quimby, *Radiology* **27,** No. 2, 196 (1936).
20. A. Reisner, *Ergeb. d. med. Strahlenforsch* **6,** (1933)
21. Studien uber die Kumulative Sirkung der Rontgenstrahlung bei Franktionierung. *Acta Radiol. Suppl. IV* (1944). Reported in R. D. Evans, *Problems Associated with the Transportation*

of Radioactive Substances. National Academy of Sciences, Washington, D.C. (1951).
22. J. R. Raper and K. K. Barnes in *Biological Effects in External Beta Radiation* (Edited by R. E. Zirkle) p. 77. McGraw-Hill, New York (1951).
23. C. A. Sondhaus, *The Ratio of Lung Beta Dose to Whole Body Dose During Given Intervals After Atomic Detonation* USNRDL (Confidential) (1952)
24. R. C. Bolles and N. E. Ballou, *Calculated Activities and Abundances of* U^{235} *Fission Products.* USNRDL (1956).
25. H. H. Rossi and R. H. Ellis, Jr., *Nucleonics* **7,** No. 1, 18 (1950); **7,** No. 2 (1950).
26. H. H. Rossi and R. H. Ellis, Jr., *Amer. J. Roentgen.* **67,** No. 6, 980 (1952).
27. E. Tochilin and P. Howland, *Interpretation of Survey Meter Data, Operation Greenhouse* (Classified) (1951).
28. F. Ellinger, *The Biologic Fundamentals of Radiation Therapy* (Translated by R. Gross). Nordeman Publishing Co., New York (1941).
29. J. R. Raper, R. E. Zirkle and K. K. Barnes in *Biological Effects in External Beta Radiation* (Edited by R. E. Zirkle) p. 1. McGraw-Hill, New York (1951).
30. G. W. Imirie, Jr. and R. Sharp, *Radiation Energy Absorbed by Human Phantoms in a Fission Fallout Field.* (Confidential) (1955).
31. G. C. Dale and J. E. Bomyer, *Dose Rates from Ground Contamination with Residual Radioactive Materials from an Atomic Explosion.* Report No. 0-35/56(X); Atomic Weapons Research Establishment (1956).
32. Project performed by Dr. W. A. Gross, Department of Radiology, Columbia University, New York.
33. A. Broido and J. D. Teresi, *Tolerance in Man to External Beta Radiation.* USNRDL-TM-4 (1954).
34. J. D. Teresi and A. Broido, *Discussion of the Ratio of Beta Surface Rep Dose to Gamma Dose Associated with Fission Product Contamination.* USNRDL-TM-2 (1954).
35. A. Broido and J. D. Teresi, *Requirements for Mass Decontamination of Personnel.* USNRDL-TR-38 (1955).

Health Physics Pergamon Press 1959. Vol. 1, pp. 390–404. Printed in Northern Ireland

PENETRATION OF WEAPONS RADIATION: APPLICATION TO THE HIROSHIMA–NAGASAKI STUDIES

R. H. RITCHIE and G. S. HURST
Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee

(*Received* 11 *November* 1958)

Abstract—Basic data necessary for the consideration of shielding against nuclear weapons radiation are presented. These include the variation of dose with distance from nuclear detonations, angular distribution of neutrons and γ-radiation, and some information on the spectrum of fast neutrons. Application of the data is made to the problem of determining the radiation dose in light frame structures. Experimental attenuation factors for typical Japanese houses are given.

1. INTRODUCTION

Some of the energy released from an atomic explosion appears in the form of γ-radiation and fast neutrons. Because of their penetrability in air, these radiations may produce severe biological damage at great distance from the point of detonation. This fact has led both military and civilian groups to consider the problem of shielding of weapons radiation. Information which is basic to the problem of determining the degree of shielding afforded by arbitrary arrangements of shielding materials will be covered in this report.

Before progressing to details of characteristics of weapons radiation, a few general remarks on the source of γ-rays and fast neutrons may be useful.[1] The γ-rays produced in the initial fission process are largely absorbed in the device itself; thus the main sources of γ-radiation are (1) the short life fission products in the fireball and (2) the nitrogen in air and material in the weapon which produce γ-rays by neutron capture.[2] Almost all of the neutrons are produced in the initial fission process.

Extensive measurements of neutron and γ-radiation have been carried out at U.S. weapons tests. In the case of γ-radiation, most of the dose measurements have been made with photographic film dosimeters, although recently much of the work has been done with chemical dosimeters. The tetrachloroethylene system[3] has been of particular interest because of its apparently negligible response to neutrons. The earliest measurements of neutrons[4] used Au and Au + Cd to determine the flux of thermal neutrons and the $S^{32}(n,p)P^{32}$ reaction to measure the flux of neutrons above approximately 2.5 MeV. More recently,[5–6] these measurements have been supplemented through the use of Pu^{239} (surrounded with B^{10}), Np^{237}, and U^{238}. The complete system provides enough information on the spectrum of neutrons to permit a reasonably accurate calculation of tissue dose.

This report will present information, obtained through the use of these two methods of dosimetry, which will be useful in considerations on the shielding of atomic weapons radiation. For this purpose exact knowledge of the "dose" yield as a function of weapons design is not essential and will not be covered.

The information which is basic to shielding consideration—namely: (1) penetration of radiation in an air-over-ground geometry; (2) energy spectral distributions; and (3) angular distribution of weapons radiations—will be illustrated for "nominal bombs" (10–20 kT). Finally, the application of these data to a particular shielding problem, that of determining the shielding afforded by typical Japanese houses, will be discussed.

2. DOSE VS. DISTANCE

To consider the variation of dose with distance from the point of detonation, let R be the

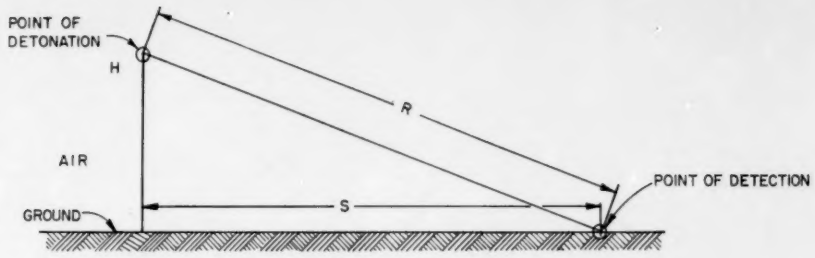


FIG. 1. Diagram illustrating the geometry of field measurements.

slant range from the point of detonation to the point of detection, H be the height of burst, and S the projection of R on a horizontal plane (Fig. 1). The particular geometry is air-over-ground with the point of detection on or very close to the ground.

Experience[7] has shown that the dose-distance relationship for γ-rays and neutrons can be represented by

$$D(R) = \frac{G_0 \exp(-R/L)}{R^2} \tag{1}$$

where G_0 depends on weapon yield as well as design, and L is the relaxation length for the radiation in question. Even for the conditions specified in Fig. 1, the relaxation length will depend on the energy spectrum of the radiation (hence weapons design) and air density. Since we are not attempting to correlate radiation dose with weapons design, we consider only the latter effect and set

$$L = \frac{\rho_0}{\rho} L_0 \tag{2}$$

where ρ is the air density and ρ_0 and L_0 refer to the values of these quantities at a density of 1.29 g/l.

Typical dose–distance data for γ-rays are shown in Fig. 2. The log of the quantity $R^2D(R)$ is plotted against R yielding a straight line. Using the known air density at the time these data were obtained, the value of $L_{0\gamma}$ for this case is 294 yd. The value $G_{0\gamma}$/kT is quoted for a nominal device and is equal to 4.1×10^9 rads-yd^2/kT in this case. To illustrate the variation in these parameters with weapons design, WILSON[8] gives $L_{0\gamma} = 350$ yd and $G_{0\gamma}/\text{kT} = 1.75 \times 10^9$ yd^2-r/kT for the Hiroshima and Nagasaki weapons. GLASSTONE[1] gives $L_{0\gamma} = 338$ yd and $G_{0\gamma}/\text{kT} = 1.4 \times 10^9$ yd^2-r/kT for a nominal weapon.

As shown in Fig. 3, equation (1) holds equally well for fast neutron dose. Figure 3 is plotted in the same way as Fig. 2 except in the case of fast neutrons the flux as a function of distance $F(R)$ above the thresholds of the detectors is presented. The dose curve $D(R)$ is derived from the flux measurements as follows:[5]

$$D = [1.0(N_{\text{Pu}} - N_{\text{Np}}) + 2.4(N_{\text{Np}} - N_{\text{U}}) + 3.0(N_{\text{U}} - N_{\text{S}}) + 3.7N_{\text{S}}] \times 10^{-9} \tag{3}$$

where D is the tissue dose in rads, N_{Pu} is the neutron flux above 5 keV (assuming 2.2 g/cm^2 elemental B^{10}), N_{Np} is the neutron flux above 0.75 MeV, N_{U} is the neutron flux above 1.5 MeV, and N_{S} is the neutron flux above 2.5 MeV. The thermal neutron contribution to the dose is not included in this formula since for all cases measured it is negligible compared with the fast component. For the case shown in Fig. 3, $L_{0_n} = 212$ yd for all the flux curves and the dose curve. This illustrates the important empirical observation that the fast neutron energy spectrum does not appear to change with distance for the geometry considered in Fig. 1. This fact has been observed to hold true for a variety of weapons fired under a variety of conditions, including changes in the value of the burst height H. The existence of this apparent constancy in spectrum is analogous to the existence of an equilibrium spectrum for deep γ-ray penetration problems[9] and is probably explained by similar considerations, i.e. the pertinent neutron cross-sections show a generally increasing behavior with decreasing energy so that the most energetic components of the

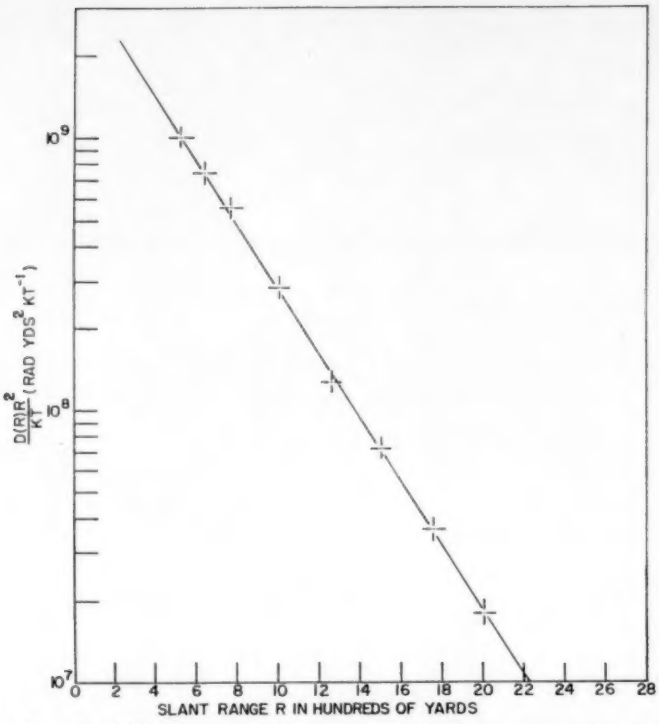


FIG. 2. γ-Air dose vs. slant range for a typical nuclear detonation.

neutron spectrum control the penetration of the beam. Consequently, the low energy portions of the spectrum are fed by neutrons slowing down from higher energy groups but are depleted with a relatively greater probability than the higher energies because of the larger cross-sections.[10] The value of G_{0_n}/kT for the case illustrated in Fig. 3 is 1.5×10^{10} rads-yd^2/kT.

3. ANGULAR DISTRIBUTION OF WEAPONS RADIATION

Information on the characteristics of ionizing radiation at large distances in air from nuclear weapons was limited mainly to air dose data until Operation Plumbbob in Nevada in 1957. The need for a more detailed description of the radiation field has been clearly felt; at least one attempt to measure the energy spectrum of γ-rays at considerable distance from a nuclear device[11] has been made, and some information on the fast neutron spectrum has been obtained with threshold detectors.[7] For purposes of shielding against weapons radiation, a knowledge of the angular distribution of these radiations is essential. Accordingly, plans were made late in 1956 for a determination of this quantity during Operation Plumbbob.[12]

(a) *Collimators*

The arrangement selected for the purpose of defining the range of directions from which radiations are accepted is shown in Figs. 4(a) and 4(b). The collimators consist of large cylindrical tanks with conical apertures. As mentioned above, the detectors, which consist either of the γ-ray chemical dosimeters or the neutron threshold detectors, are placed at the apex of the conical openings. The basic collimators were designed with an opening having a half-angle equal to 22.5°. Conical inserts were designed so that the opening could be reduced to 15° or 10°. The tanks were filled entirely with water for use in neutron measurements, Fig. 4(a). For the γ-ray application,

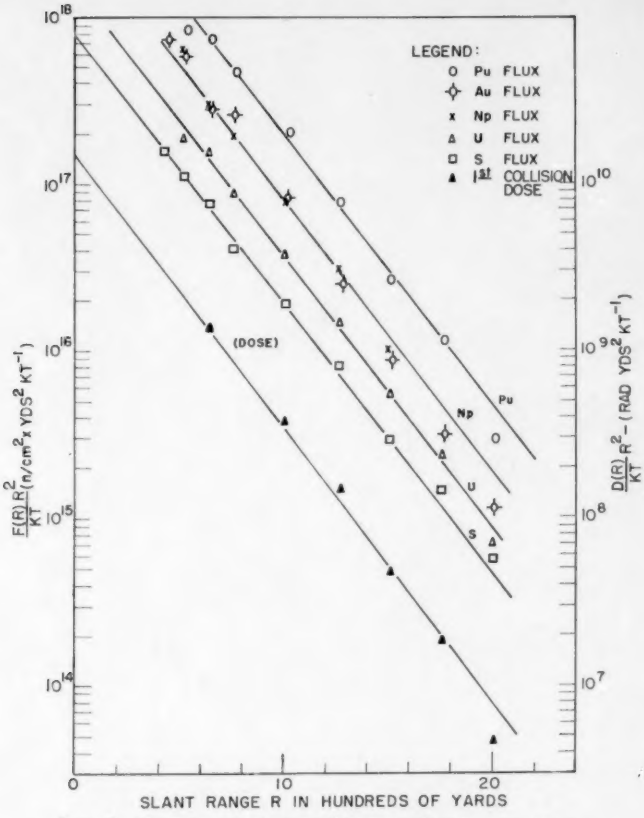
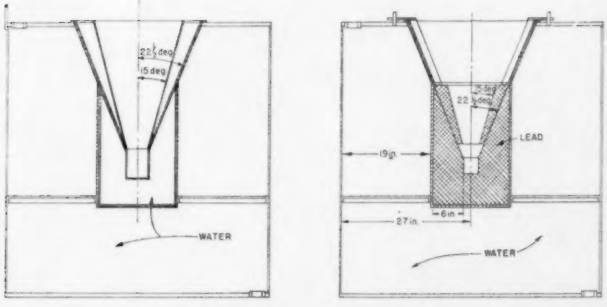


FIG. 3. Neutron air dose and flux vs. slant range for a typical nuclear detonation.



NEUTRON COLLIMATOR (a)

GAMMA COLLIMATOR (b)

FIG. 4. Diagram showing the construction of neutron and γ-ray collimators.

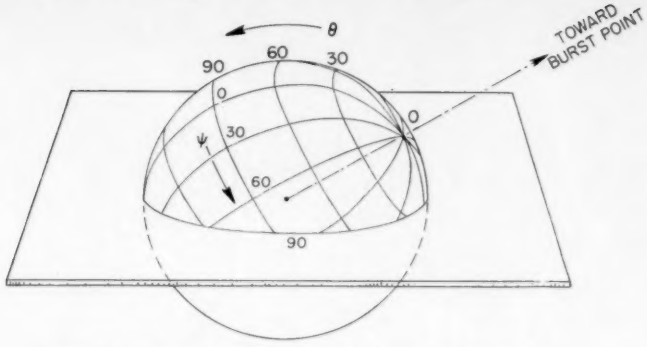


FIG. 5. Diagram illustrating the co-ordinates θ' and Ψ'' used in presenting angular distributions.

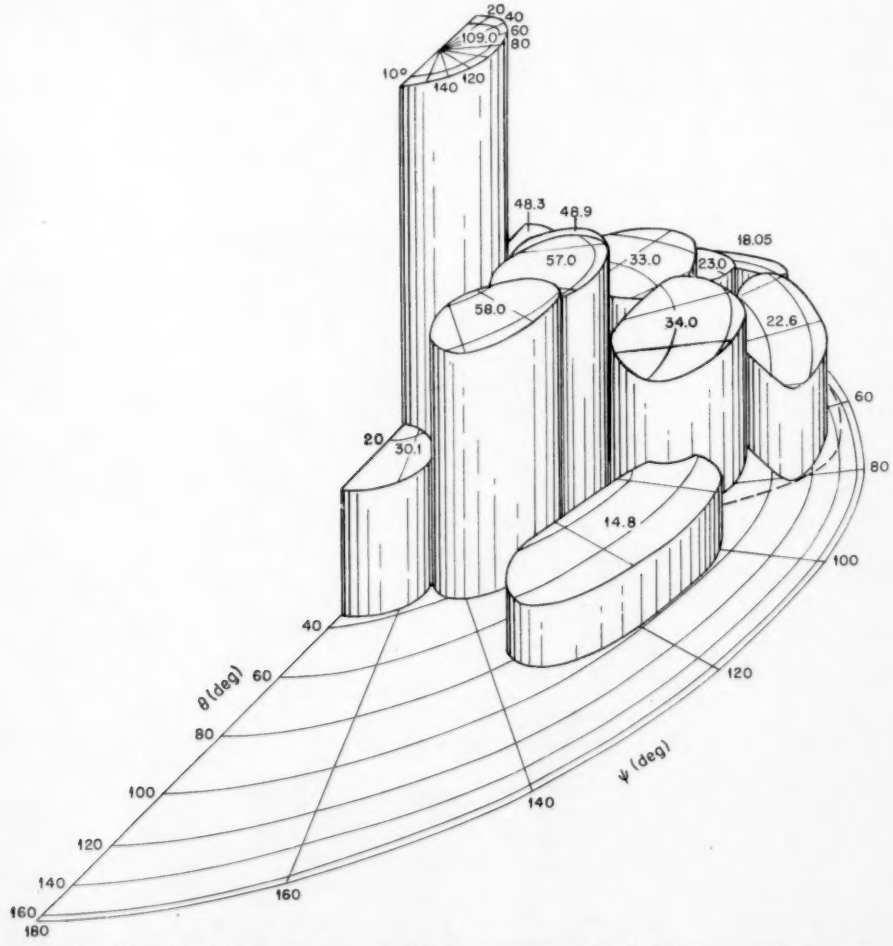


FIG. 6. Angular distribution of neutrons above plutonium threshold for a typical nuclear detonation. The heights of the columns are proportional to the flux entering the collimators oriented as indicated by the location of the columns.

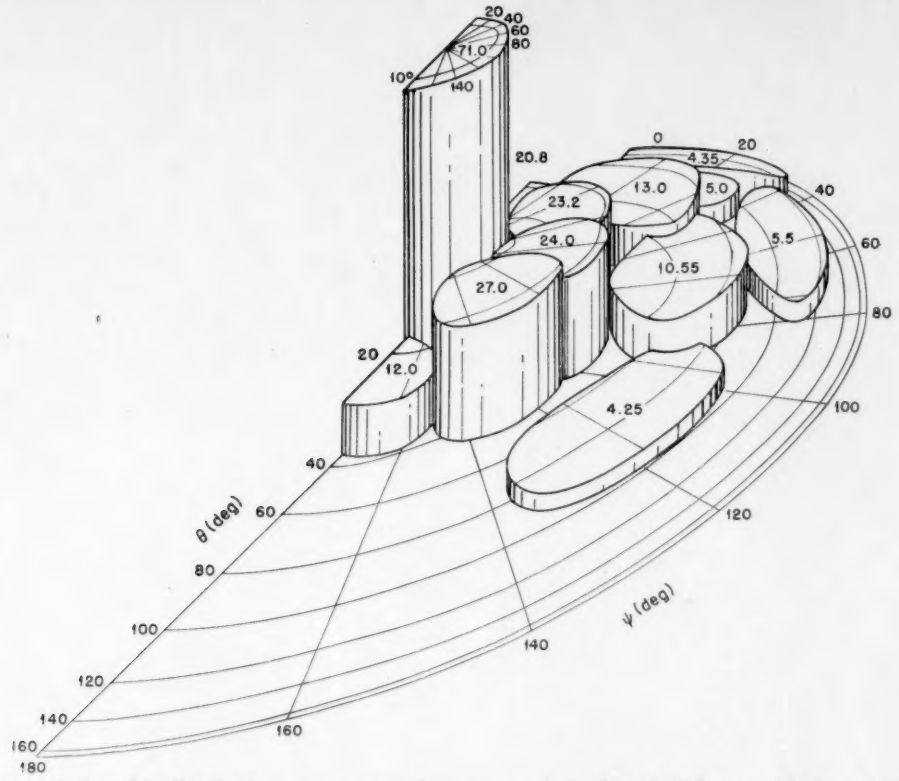


FIG. 7. Angular distribution of neutrons above neptunium threshold for a typical nuclear detonation. The heights of the columns are proportional to the flux entering the collimators oriented as indicated by the location of the columns.

part of the water around the apex of the cone was replaced by a lead cylinder, Fig. 4(b), which is an efficient attenuator of γ-radiation. The water outside the lead served to attenuate the neutrons that would strike the lead and thus minimized the generation of inelastic γ-rays therein. The overall attenuation factor of the collimators for radiation incident from directions outside the cone of acceptance was estimated to be at least 10^{-3}. Laboratory tests made with artificial sources showed that the angle of acceptance of radiation was only slightly larger than the angular opening of the collimators due to wall scatter of radiation.[13] The data are corrected for this effect.

(b) *Measurements*

For field measurements the collimators were grouped in clusters at various distances from the burst point and were distributed in direction so as to cover as much as possible of the whole sphere of directions at each distance. The system of co-ordinates used to describe the data obtained is presented in Fig. 5.

Since the collimators subtend an appreciable solid angle, it is desirable to have an indication of the extent of coverage. At a specified orientation of a collimator, the opening subtends a circular cap on a sphere whose center is at the apex of the collimator. These circles are mapped on a plane co-ordinate system and the mapping is chosen so that equal areas on the sphere map into equal areas on the plane system. Representative three-dimensional plots of Pu, Np, U and S activation fluxes are presented in Figs. 6–9, respectively. A given

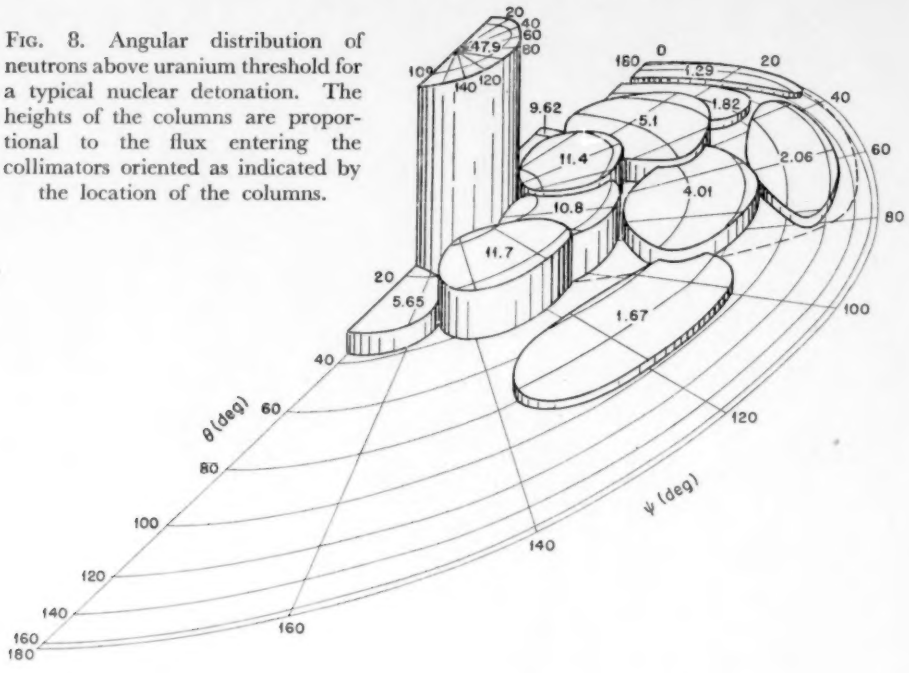


FIG. 8. Angular distribution of neutrons above uranium threshold for a typical nuclear detonation. The heights of the columns are proportional to the flux entering the collimators oriented as indicated by the location of the columns.

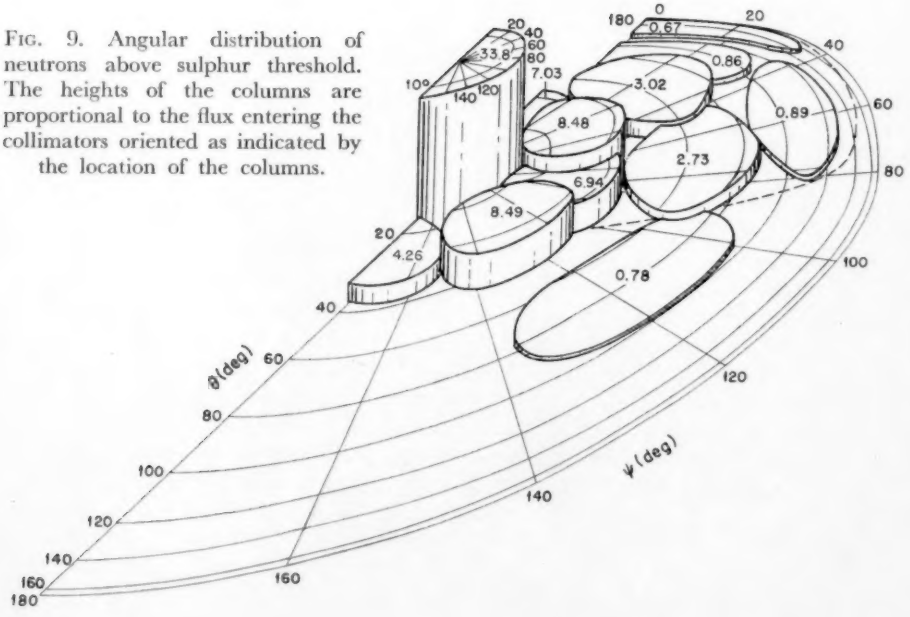


FIG. 9. Angular distribution of neutrons above sulphur threshold. The heights of the columns are proportional to the flux entering the collimators oriented as indicated by the location of the columns.

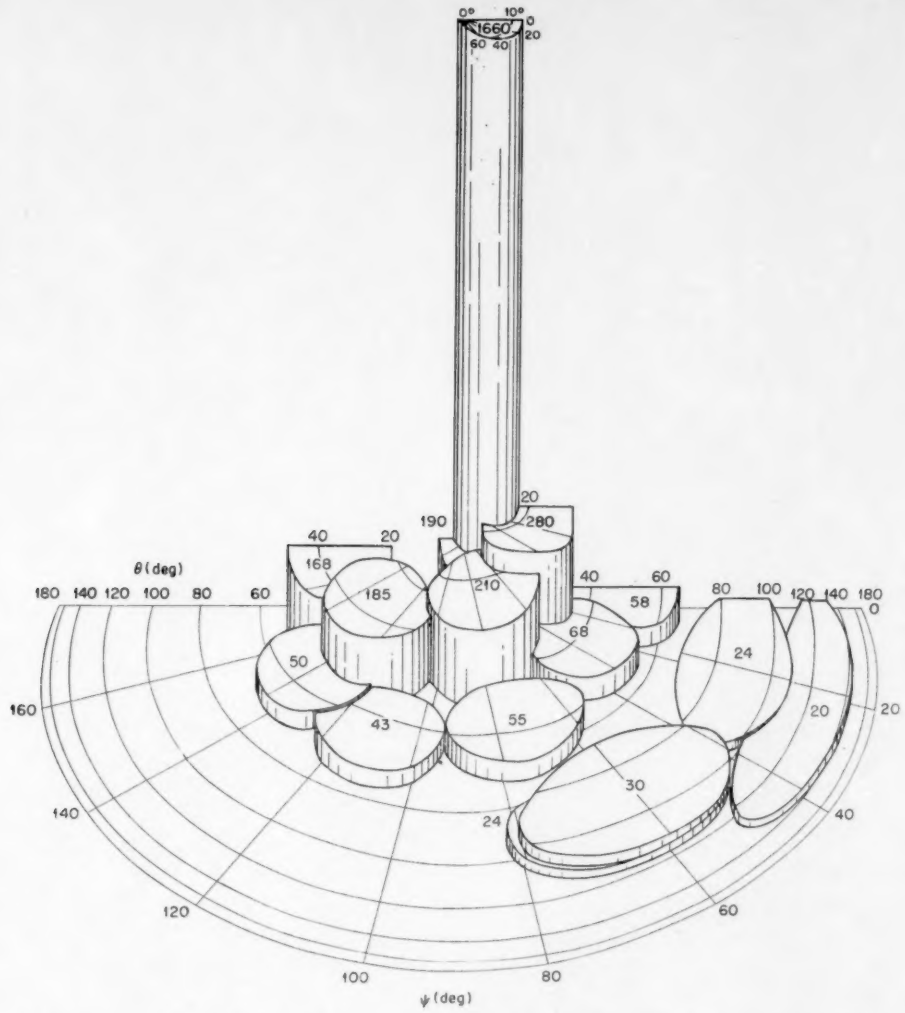


FIG. 10. Angular distribution of γ-rays for a typical nuclear device. The numbers on the columns are proportional to the doses entering the collimators indicated.

column height in these figures is proportional to the flux entering the collimator whose angular aperture coincides with the column base. Fig. 10 shows a similar plot of the γ-dose in another event. In this case a given height in the figure is proportional to the dose entering the collimator. In both cases the distributions are normalized to that which would be indicated by 15° half-angle collimators. The overall normalization is arbitrary.

In the several events for which angular distribution data were obtained in Operation Plumbbob, the angular distributions of both neutrons and γ-rays were observed to be rather insensitive to the type of weapon and to the distance from the burst points. Figs. 11 and 12 show, respectively, the best estimates of angular distributions for neutrons and γ-rays obtained. In each case the per cent of air dose received by a 15° collimator is plotted as a function of θ

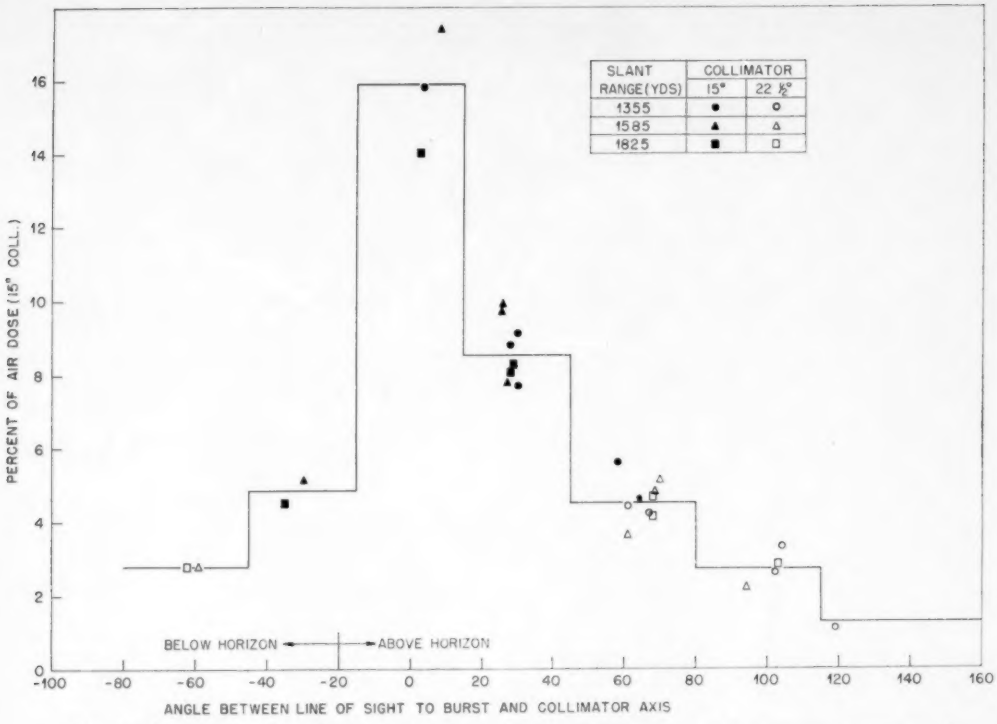


FIG. 11. Representative plot of neutron angular distribution for nuclear weapons measured. Experimental results have been grouped into two categories: (1) those above the horizon (corresponding to $0° < \Psi < 90°$), and (2) those below the horizon (corresponding to $90° < \Psi < 180°$).

(Fig. 5). Variations in the azimuthal angle Ψ have been neglected except for the division of the collimators into two groups: those whose axes point toward the ground and those whose axes are oriented above the horizon, i.e. positive θ values include data in the range $-90° \leq \Psi \leq 90°$ and negative values of θ include the ranges $90° \leq \Psi \leq 180°$ and $-180° \leq \Psi \leq -90°$. Doses received by $22\frac{1}{2}°$ collimators have been normalized to the 15° collimator response.

These data may be employed directly to estimate the dose which would be received by individuals shielded by irregularities in the terrain, e.g. ditches, foxholes, etc. They may be used to estimate the effective input source for calculating the shielding by structures of prompt radiation from weapons. These data will be compared with calculations of angular distributions when the latter become available.

4. APPLICATION TO DETERMINATION OF DOSE RECEIVED BY RESIDENTS OF HIROSHIMA AND NAGASAKI

The health physics profession is concerned with the protection of man from the deleterious effects of ionizing radiation. Limits of exposure to these radiations can best be determined from a study of humans under conditions which permit accurate measurement of radiation dose and medical effects. The group of exposed individuals in Hiroshima and Nagasaki presents a unique opportunity for a study of the medical response of a large number of humans to radiation. A long-term study of medical effects in this group is in progress in Japan at the Atomic Bomb Casualty Commission, operated by the U. S. National Academy of Sciences, National Research Council.[14] The program is conducted in co-operation with the National

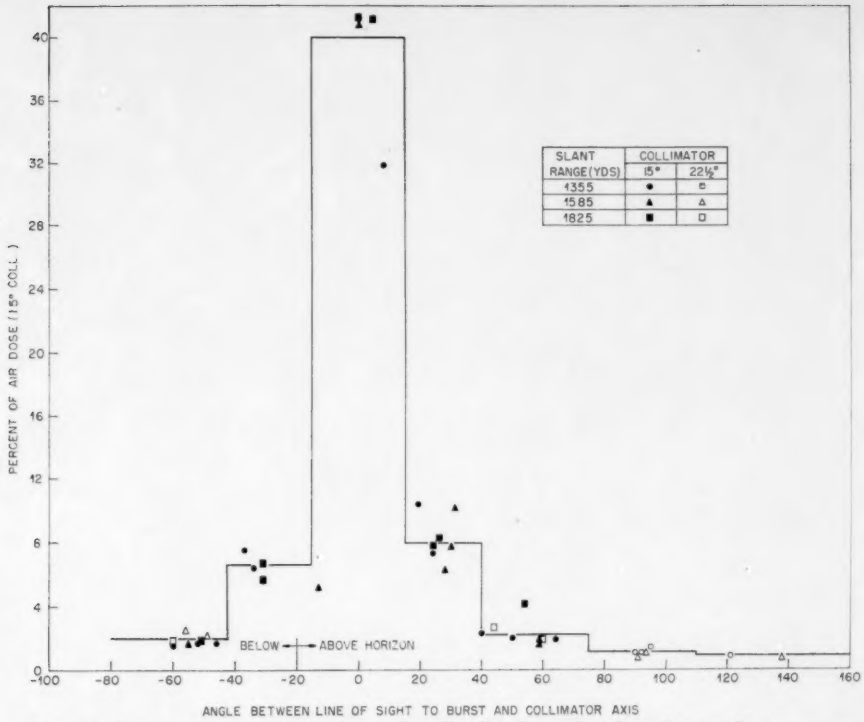


FIG. 12. Representative plot of angular distribution of γ-dose for nuclear weapons measured. Experimental results have been grouped into two categories: (1) those above the horizon (corresponding to $0° < \Psi < 90°$), and (2) those below the horizon (corresponding to $90° < \Psi < 180°$).

Institute of Health of the Ministry of Health and Welfare of the Japanese Government, with participation of interested Japanese scientists.

A critical study of the dose received by these individuals has just begun in the United States. It appears that a body of accurate data on the location of and shielding configuration around many persons exposed in the two cities now exists and continues to accumulate. The group of exposed persons which is of greatest interest for immediate study is the group located in light residential structures at the time of the bombing. Not only do a large fraction of the people fall in this category, but the attenuation factors of light structures are easier to establish than for more massive buildings. The experimental studies described in this paper have definitely decreased the uncertainty in the attenuation factors of structures of this type for radiation. There is certainly more to be accomplished in this area, but it now appears that the prime uncertainty in the determination of dose to individuals lies in establishing the air dose. The neutron and γ-ray air dose distributions for both Hiroshima and Nagasaki (Figs. 13 and 14) have been estimated by YORK[15] using all available weapons effects information. The dotted curves show the range of estimated probable errors associated with each radiation. The averages of the dose curves agree to within 20 per cent in the case of γ-rays and to within 30 per cent in the case of neutrons with similar estimates of HARRIS.[16] The earliest estimates of WILSON[8] for the γ-doses agree to within 30 per cent over the most interesting distances. WILSON does not estimate the neutron dose to the Japanese. In spite of this good agreement between independent estimates there is still considerable uncertainty in the air dose, as can be clearly seen from Figs. 13 and 14.

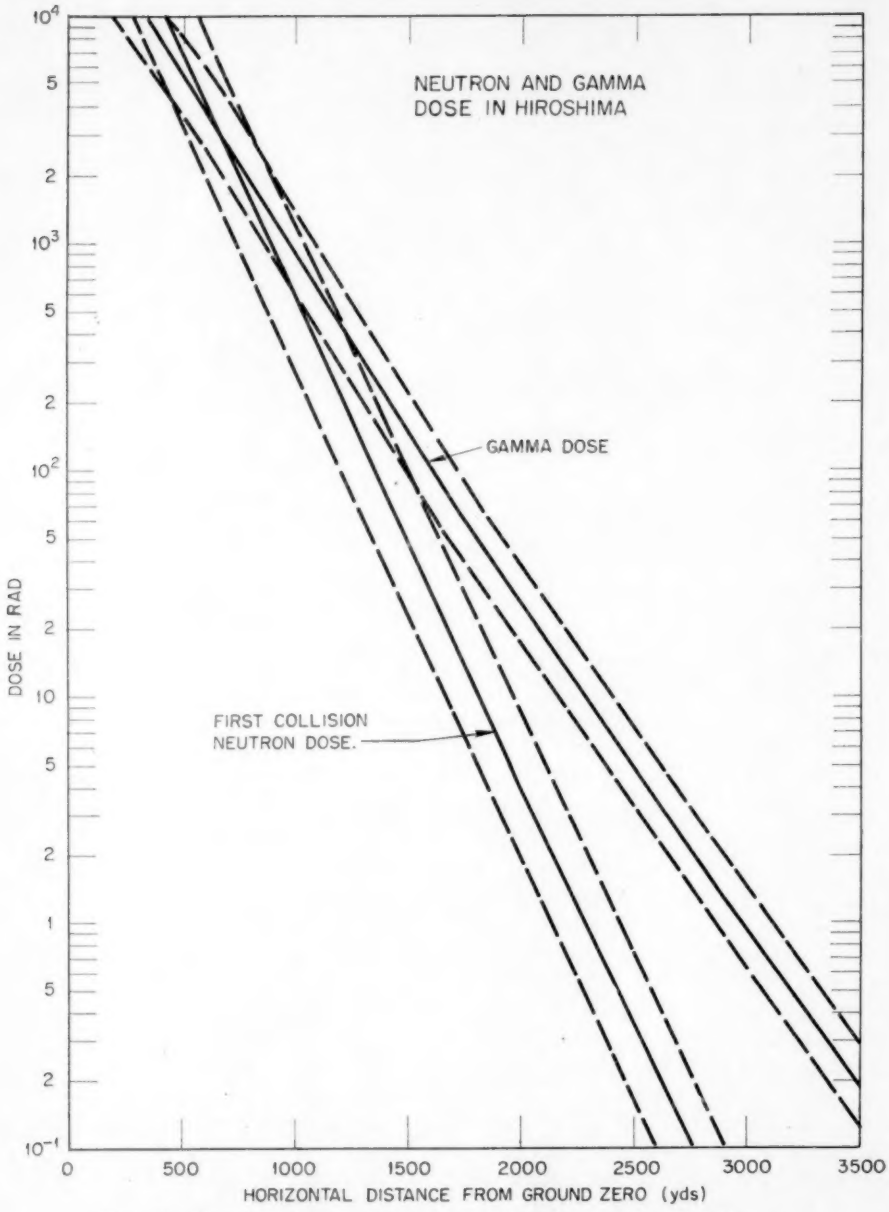


FIG. 13. YORK's estimate of neutron and γ-air dose in Hiroshima as a function of horizontal distance from ground zero.

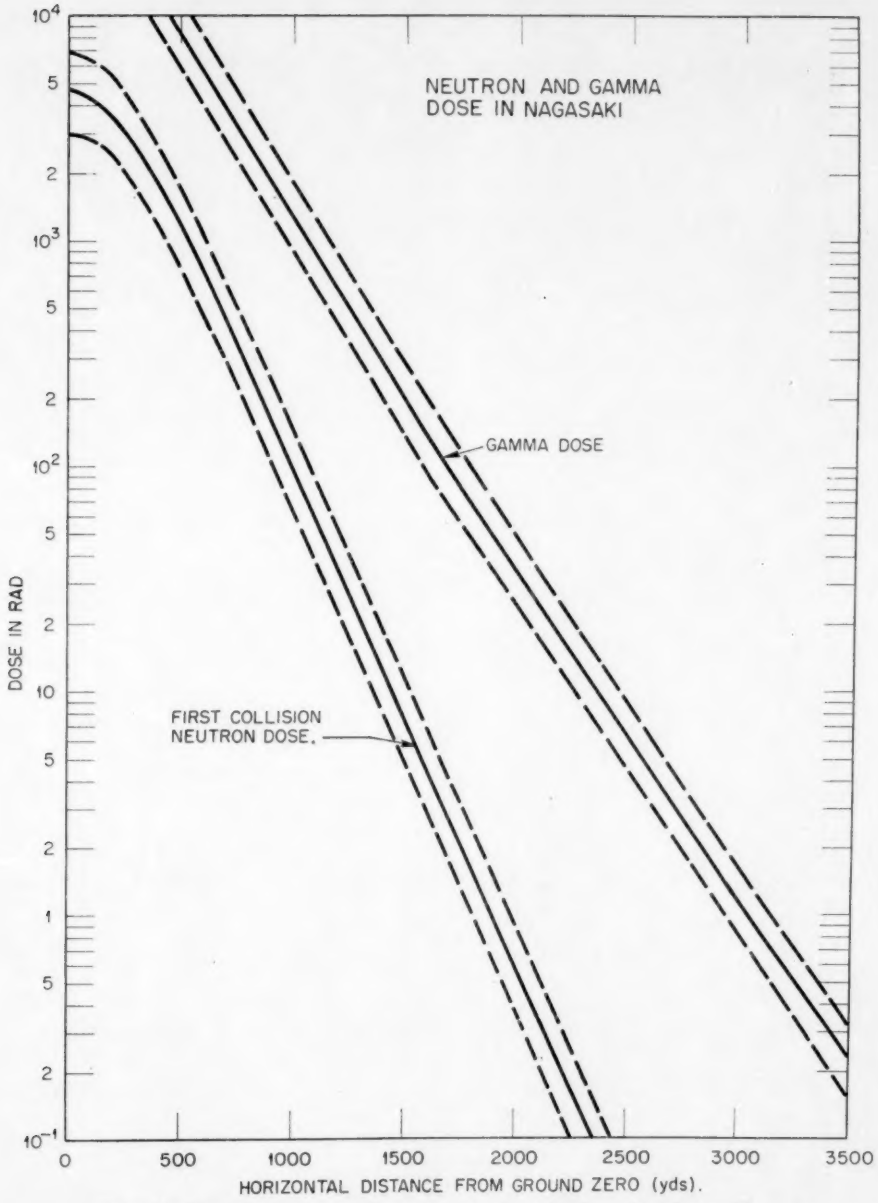


FIG. 14. YORK's estimate of neutron and γ-air dose in Nagasaki as a function of horizontal distance from ground zero.

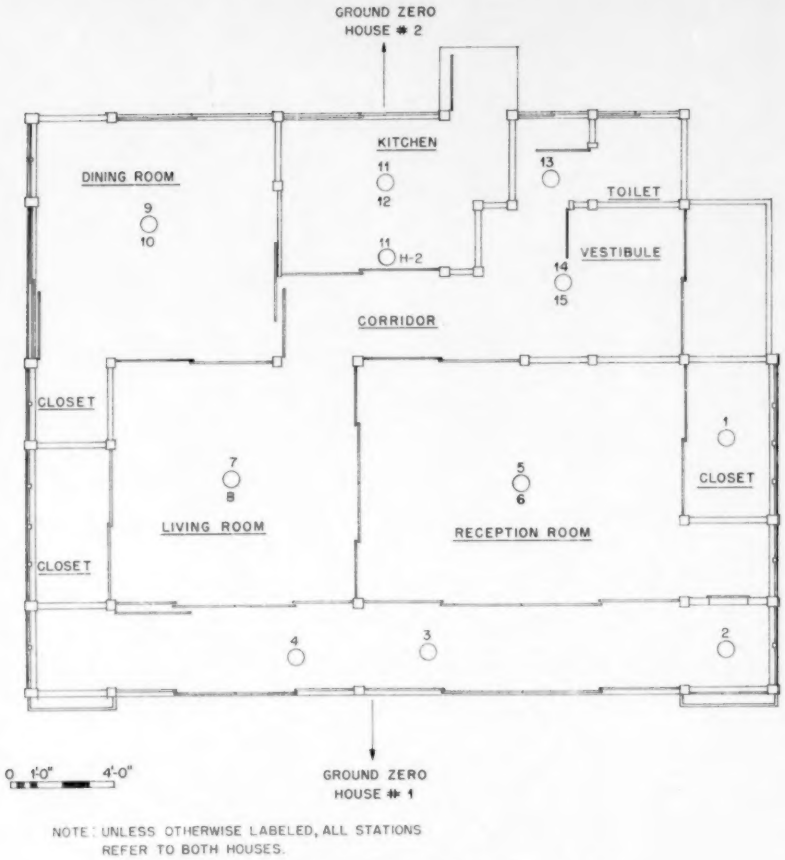


FIG. 17. Plan of house showing dosimetry stations.

The attenuation of two light frame structures was determined in the Plumbbob operation. The houses were identical in construction and were oriented so that the open (veranda) side of one house faced ground zero, while the veranda side of the other house faced in the opposite direction. Fig. 15 shows a view of the veranda side of the house while Fig. 16 shows the opposite side. The construction and layout of the houses were typically Japanese. The tile roof had a mean superficial density of about 9 g/cm^2 including the mud base and wood sheathing. The mud-on-bamboo walls were approximately the same superficial density, while the interior walls were 11 g/cm^2.

A total of 30 neutron- and 30 γ-detectors, all unshielded, were exposed in each of the houses. According to the arrangement shown in Fig. 17, half of the detectors were exposed at a height of approximately 42 in. above the floor while the remainder were placed 84 in. above the floor. The results obtained for the neutron- and γ-attenuation factors are shown in Figs. 18 and 19, respectively. In both cases it is seen that the dose correlates very well with a simple parameter, viz. the distance measured along the ray path from the point of entry into the house to the dosimeter (house penetration distance). An equally good correlation is found if one uses instead of the house penetration distance, the distance from the dosimeter to the outside wall of the house, measured along the projection of the ray path on the horizontal. Such correlations are not surprising in view of

FIG. 15. Veranda side of light frame structure.

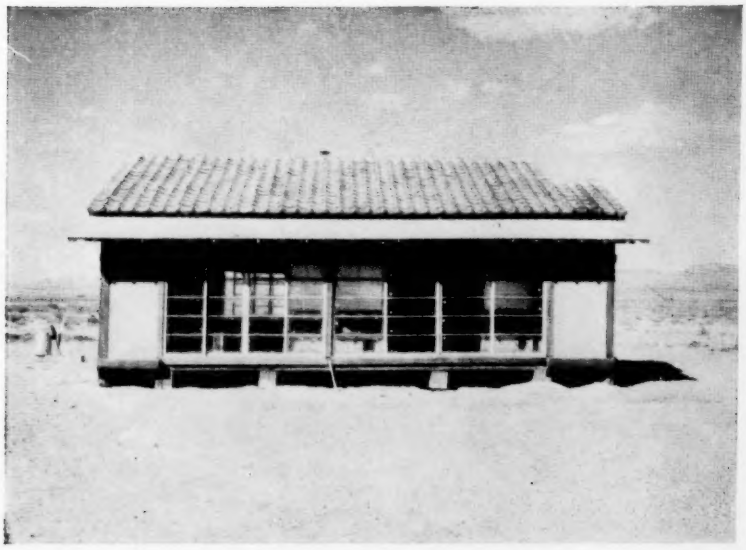


FIG. 16. North side of light frame structure.

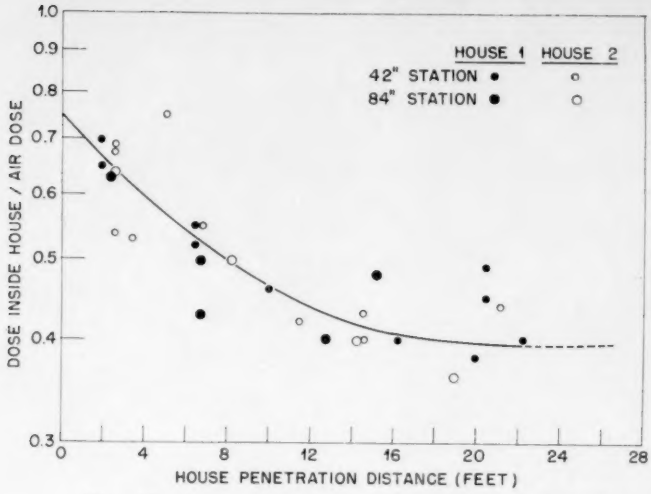


FIG. 18. Attenuation of fast neutrons by typical single story Japanese houses.

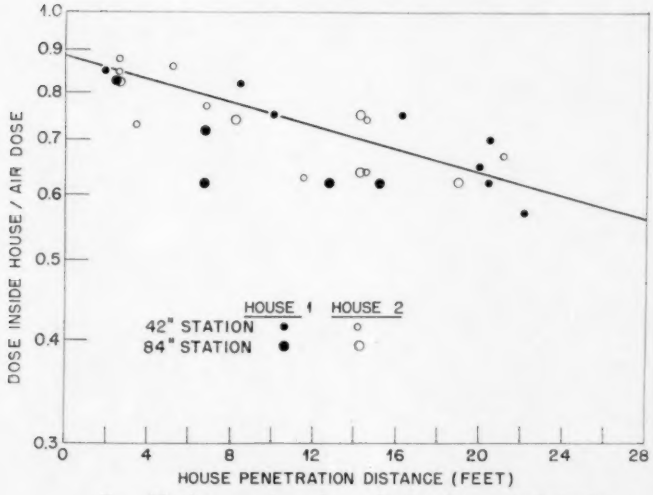


FIG. 19. Attenuation of γ-radiation by typical single story Japanese houses.

the angular distributions found experimentally. Crudely speaking, one may regard a house as a shell with holes (windows) in it. If there were no holes, the radiation level would be fairly uniform throughout at about 0.7 and 0.4 of the outside level for γ-rays and neutrons, respectively. The large open spaces in the walls permit some of the radiation to enter unattenuated, and the farther one moves from the windows, the smaller is this contribution.

The scatter in the dose data inside the houses is approximately from 10 to 15 per cent and is well within the uncertainty in the air dose values in both Hiroshima and Nagasaki. Further house attenuation tests are underway for the purpose of investigating the effect of varying house size, orientation and mutual shielding of one house by another.

One may now arrive at a tentative dose estimate for people exposed in Japanese homes by the following procedure: (1) determine the air dose by means of YORK's air–dose curves and the known location of the survivors; (2) assign a house attenuation factor by reading the house penetration distance from the ABCC shielding records. Clearly, the results obtained in this way are very crude. In order to reduce the uncertainty in the dose it will be necessary to (1) establish more accurate the radiation yield of the Hiroshima and Nagasaki devices, and (2) obtain information on radiation attenuation of more general house shielding configurations. This can best be accomplished by a combination of experimental and calculational approaches, using the measured angular distributions as input information.

Acknowledgments—The authors wish to acknowledge the participation of the following individuals in this work: Health Physics Division, Oak Ridge National Laboratory: E. T. ARAKAWA, J. A. AUXIER, C. W. CALDWELL, J. S. CHEKA, F. J. DAVIS, J. A. HARTER, P. N. HENSLEY, W. W. OGG, P. W. REINHARDT, C. C. SARTAIN, E. B. WAGNER, B. T. WALTERS. U. S. Air Force, School of Aviation Medicine: H. M. BORELLA, L. C. LOGIE, S. C. SIGOLOFF.

The loan of T. V. BLOSSER to this project from the Applied Nuclear Physics Division, ORNL, is greatly appreciated. The valuable work of Capt. E. N. YORK and Col. M. MORGAN, Air Force Special Weapons Center, is gratefully acknowledged. The wholehearted support of R. L. CORSBIE and L. J. DEAL, U.S. Atomic Energy Commission-Division of Biology and Medicine, and J. A. PICKERING, U.S.A.F. School of Aviation Medicine, throughout the course of the field work is gratefully acknowledged.

REFERENCES

1. S. GLASSTONE (Editor), *The Effects of Nuclear Weapons*. United States Atomic Energy Commission (1957).
2. P. S. MITTELMAN and R. A. LIEDTKE, *Nucleonics* **13,** No. 5, 50 (1955).
3. S. C. SIGOLOFF, *Nucleonics* **14,** No. 10, 54 (1956).
4. W. A. BIGGERS and F. WADDELL, WT-9004 (Classified). U. S. Atomic Energy Commission (1957).
5. G. S. HURST, J. A. HARTER, P. N. HENSLEY, W. A. MILLS, M. SLATER and P. W. REINHARDT, *Rev. Sci. Instrum* **27,** 153 (1956).
6. P. W. REINHARDT and F. J. DAVIS, *Health Physics* **1,** 169 (1958).
7. S. GLASSTONE (Editor), *The Effects of Nuclear Weapons*. U. S. Atomic Energy Commission (1957); P. S. HARRIS *et al.*, ITR-1167. (Classified). U. S. Atomic Energy Commission (1955).
8. R. R. WILSON, *Rad. Res.* **4,** 349 (1956).
9. U. FANO, *Nucleonics* **11,** No. 8, 8 (1953); **11,** No. 9, 55 (1953).
10. S. S. HOLLAND, JR. and P. I. RICHARDS, *J. Appl. Phys.* **27,** 1042 (1956).
11. Private communication with R. WATT, Los Alamos Scientific Laboratory.
12. G. S. HURST, *et al.*, WT-1504 (Classified). U.S. Atomic Energy Commission (1958).
13. J. A. AUXIER and T. V. BLOSSER. To be published
14. A. W. OUGHTERSON and S. WARREN (Editors), *Medical Effects of the Atomic Bomb in Japan*. McGraw-Hill, New York (1956).
15. E. N. YORK in communication from M. MORGAN, AFSWC, to G. S. HURST, ORNL, ORNL-CF-57-11-144 (1957).
16. P. S. HARRIS, Los Alamos Scientific Laboratory Report LA-1987 (Classified) (1955); J. V. NEEL and W. J. SCHULL, *The Effect of the Atomic Bomb on Pregnancy Termination in Hiroshima and Nagasaki* p. 51. NAS-NRC Publication 461 (1956).

Health Physics Pergamon Press 1959. Vol. 1, pp. 405–408. Printed in Northern Ireland

RADIOASSAY OF URANIUM AND PLUTONIUM IN VEGETATION, SOIL AND WATER

E. L. GEIGER*
Savannah River Plant
E.I. du Pont de Nemours and Company
Aiken, South Carolina

(*Received* 24 *October* 1958)

Abstract—A method is discussed for the separation of uranium and plutonium from vegetation, soil and water. The method is based on the extraction of uranium and plutonium from 4 to 6 N nitric acid into 50% tri-*n*-butyl phosphate in *n*-tetradecane diluent. Uranium and plutonium are recovered together with sufficient reduction in total solids to allow α-counting and pulse height analysis. Data from several hundred "spiked" samples to which uranium and plutonium were added indicate a nearly equal recovery of uranium and plutonium. Average recoveries are 76 ± 14 per cent for vegetation, 76 ± 16 per cent for soil, and 82 ± 15 per cent for water. The procedure is designed for samples that may be collected and analyzed for radioactivity as a part of a health physics regional monitoring program.

INTRODUCTION

THE environs of atomic energy plants are monitored for uranium and plutonium for control purposes. Samples normally collected include vegetation, soil and water. The problem of determining the α-activity of these samples is principally one of isolating uranium and plutonium from the sample bulk so that only minute amounts of solid material are associated with the uranium and plutonium. α-Counting methods are accurate and sensitive provided self-absorption effects, due to the presence of solids, are negligible.

An ether extraction method has been used to separate uranium and plutonium from the bulk of solid material in the analysis of vegetation, soil and water samples.[1,2] Ether, however, is very flammable and sometimes produces a vigorous reaction when in contact with nitric acid solution. Tri-*n*-butyl phosphate (TBP) and other extractants have been evaluated by other investigators and compared with diethyl ether.[3,4,5] A purer product has been obtained with TBP in hydrocarbon diluent than with diethyl ether.[5] The primary advantage, however, is that TBP is less hazardous. This paper discusses an investigation to adapt the TBP method to the determination of uranium and plutonium in samples collected for health physics monitoring.

DISCUSSION

To apply the TBP extraction method to vegetation, soil and water, sample preparation procedures were developed to recover uranium and plutonium from these samples in a nitric acid solution. Approximately 5 N nitric acid free of fluorides was required to permit optimum, simultaneous extraction of uranium and plutonium into TBP.

Recovery of uranium and plutonium in acid solution

Two methods of sample preparation for recovery of uranium and plutonium from vegetation were investigated. One method provided complete dissolution of the sample by a combination of complete ashing of the organic material, and destruction of siliceous material by heating with a mixture of nitric and hydrofluoric acids, followed by a final dissolution of the residue in nitric acid. The other method

* Present address: Reynolds Electrical and Engineering Company, 931 South Main Street, Las Vegas Nevada.

consisted of leaching the ashed vegetation with a nitric acid solution.

Vegetation grown in an environment that was contaminated with uranium and plutonium was used to compare the two methods. The leaching method proved to be more reliable and provided higher recoveries in the TBP extraction step. This method was found to be adequate for all types of vegetation routinely sampled.

Soil samples were heated in a muffle furnace to destroy organic material, treated with an $HF–HNO_3$ solution to destroy siliceous material, and then leached with an aluminum nitrate–nitric acid solution. Attempts to leach the soil without the $HF–HNO_3$ treatment yielded low recoveries. Satisfactory recoveries were obtained after the $HF–HNO_3$ treatment provided the HF was completely evaporated.

Water samples were acidified with nitric acid and evaporated to a small volume. The solution was adjusted to 4–5 N nitric acid and filtered or centrifuged to remove the insoluble material. Attempts to dissolve the insoluble portion by heating in a muffle furnace at 600°C, followed by $HF–HNO_3$ digestion, resulted in decreased recoveries of plutonium. The recovery obtained by simply leaching the insoluble residue was adequate; therefore, complete dissolution of the sample was not required.

Extraction of U *and* Pu

The literature contains a large volume of data on the continuous, countercurrent separation of kilogram quantities of uranium and plutonium from megacurie quantities of fission products.[3] Much of these data, however, are not directly applicable to the analyses of health physics samples in which microgram and submicrogram quantities of uranium and plutonium are separated from gram quantities of other salts dissolved in the acid solution. The low acid, high salt conditions used in the separation of kilogram quantities are not considered ideal for the analysis of vegetation, soil and water. Although addition of salting agents increased the recovery of uranium and plutonium, the increased amount of salt recovered with the uranium and plutonium interfered with the α-pulse-height analysis. Higher acid concentrations with less salting agent provided cleaner planchets and satisfactory recoveries of uranium and plutonium.

A 50% solution of TBP in *n*-tetradecane diluent was selected as the extractant. Although higher concentrations of TBP provided higher extraction coefficients, poor phase separations at higher TBP concentrations were objectionable.

Sulphate and phosphate did not seriously interfere in this method; however, the presence of hydrofluoric acid sharply decreased the recovery of plutonium. Addition of aluminum nitrate or boric acid improved the recovery but did not completely eliminate the fluoride interference; therefore, solutions that contained hydrofluoric acid were evaporated to dryness prior to the extraction step.

Nitric acid was partially extracted into the TBP layer and hindered the simultaneous back extraction of uranium and plutonium into water. The TBP layer was stripped several times with water to reduce the nitric acid concentration in the TBP layer and to recover all of the uranium and plutonium. This technique has not been found to be prohibitively time consuming or inconvenient.

After purification by TBP extraction, the sample was mounted on a stainless steel planchet and counted for α-radiation. If the gross α-count exceeded a preset level, the sample was pulse-height analyzed to identify as well as to estimate the percentage of the alpha emitters.

RESULTS OBTAINED

Vegetation, soil and water samples were spiked with uranium as well as plutonium and were analyzed along with routine samples. Over a period of seventeen consecutive months 360 samples spiked with uranium and 362 samples spiked with plutonium were analyzed. The average recovery was determined monthly. Averages and standard deviations were calculated for the entire seventeen month period. These data are summarized in Table 1. Uranium and plutonium were nearly equally recovered, the averages being 76 per cent for vegetation, 76 per cent for soil, and 82 per cent for water. The standard deviation ranged from ± 12 per cent to ± 17 per cent. The fact that these data were obtained under conditions of

routine analysis by nontechnical personnel trained as laboratory technicians enhances the value of the data in demonstrating procedure reliability.

Table 1. *Recovery* (%) *of uranium and plutonium from spiked samples*

	Vegetation		Soil		Water	
	U	Pu	U	Pu	U	Pu
Recovery (%)						
October	78	75	—	—	59	67
November	85	75	—	—	85	92
December	84	74	60	80	85	82
January	—	—	71	72	81	80
February	74	82	71	73	68	78
March	84	73	79	81	76	82
April	—	—	82	76	92	83
May	—	—	79	80	91	84
June	75	64	88	83	82	70
July	76	68	84	84	81	89
August	—	—	58	58	82	88
September	69	72	75	57	81	83
October	85	97	78	84	77	81
November	73	68	87	74	81	86
December	81	69	75	73	83	87
January	91	84	84	74	86	84
February	85	69	87	80	82	85
Total no. of samples	75	78	123	122	162	162
Grand Avg (%)	79	73	76	77	81	83
Standard deviation (%)	12	16	17	16	15	15

PROCEDURES

Preparation of samples

Vegetation. Cut oven-dried vegetation into small pieces and weigh 10.0 g into a 150 ml beaker. Heat the sample at 600°C, starting with a cold muffle furnace. When only white ash remains, remove the beaker from the muffle furnace and allow to cool. Carefully add 2 ml of water, then add 10 ml of 8 N HNO_3–0.5 M $Al(NO_3)_3$ solution. Cover the beaker with a watch glass and boil the solution for 5 min. Allow to cool, add 1 ml of 2 M KNO_2 solution and transfer the sample to a 100 ml centrifuge tube. Use 4 N HNO_3 to complete the transfer. Centrifuge and decant the supernate into a 125 ml cylindrical separatory funnel graduated at 30 ml. Wash the residue with 4 N HNO_3, centrifuge, and decant the wash solution to the separatory funnel. The acid normality of the combined solutions at this point is 4–6 N and the total volume should not exceed 29 ml. Proceed to the extraction procedure.

Soil. Grind 5 g of oven-dried soil with a mortar and pestle until the entire sample can pass through a 200-mesh sieve. Weigh 1.0 g of the 200-mesh soil into a 50 ml platinum crucible and heat the sample at 600°C for 4 hr. Remove the sample from the muffle furnace and allow to cool. Add 3 ml of 70% HNO_3 and 10 ml of 48% HF then stir the sample for 2–3 min with a platinum rod. Heat the sample in a 200°C sand bath until all traces of moisture are removed. Repeat this HNO_3–HF treatment being careful that the sample is completely dry before proceeding to the next step. Allow the sample to cool and then add 15 ml of 6 N HNO_3–0.25 M $Al(NO_3)_3$ solution. Cover with a watch glass and heat in the sand bath for 5 min. Allow to cool and decant the solution through a filter, such as Whatman no. 40, into a 125 ml cylindrical separatory funnel graduated at 30 ml. Leave as much of the residue as possible in the crucible and repeat the hot 6 N HNO_3–0.25 M $Al(NO_3)_3$ treatment. Allow to cool and then filter. Proceed to the extraction procedure.

Water. Place 1 l. of the sample in a 1.5 l. beaker and if basic, neutralize with nitric acid. Add 15 ml of 70% HNO_3 and evaporate to 30–40 ml. Decant the solution through a filter, such as Whatman no. 40, into a 100 ml beaker. Wash the 1.5 l. beaker, the residue and the filter with 4 N HNO_3. Evaporate the combined solution in the 100 ml beaker to 5 ml. Add 20 ml of 4 N HNO_3, cover with a watch glass, and heat for 5 min. Transfer the sample to a 125 ml cylindrical separatory funnel graduated at 30 ml. Wash the beaker with 4 N HNO_3 and transfer to the separatory funnel, being careful that the total volume in the separatory funnel does not exceed 29 ml. Proceed to the extraction procedure.

Extraction

Add 1 ml of 2 M KNO_2 to the sample in the 125 ml cylindrical separatory funnel. Dilute to the 30 ml mark with 4 N HNO_3 and stir the solution briefly. Add 30 ml of 50% TBP in *n*-tetradecane. Agitate the solution vigorously for 4 min with an air-driven stirrer. Discard the acid portion (lower layer). Wash the TBP portion with 4 N HNO_3 and again discard the acid portion. Back extract with seven 15 ml portions of distilled water, collecting

the strip solution in a 150 ml beaker. Evaporate the combined aqueous portions to 10–15 ml, then quantitatively transfer the solution to a flamed stainless steel planchet. Allow to dry under a heat lamp, flame the planchet to burn off organic residue, and count on an α-counter. Retain for pulse–height analysis if the α-count exceeds a specified level.

REFERENCES

1. R. Z. Bouton and R. W. Trevithick, *Solvent Extraction of Uranium from Samples Typical to KAPL Health Physics Laboratory*. KAPL–718 (1951).
2. M. R. Kennedy, *Health Physics Manual of Radiochemical and Chemical Analyses and Procedures*. pp. 12–15. KAPL–A–HP–3 (1957).
3. J. R. Flanary, *Geneva Conference Paper* 539 (1955).
4. T. C. Runion, *Solvent Extraction Chemistry of TBP*. CF–51–2–145 (Del) (1951). Declassified with deletions, 14 February, 1957.
5. *Solvent Evaluation for Liquid–Liquid Extraction*, KLX–1211, Kellex Corporation, New York (1951). Declassified 28 March, 1957.

Health Physics Pergamon Press 1959. Vol. 1, pp. 409–416. Printed in Northern Ireland

THE CONCENTRATION AND DISTRIBUTION OF RADIUM IN THE NORMAL HUMAN SKELETON*

A. WALTON,† R. KOLOGRIVOV and J. L. KULP
Lamont Geological Observatory (Columbia University)
Palisades, New York

(*Received* 21 *August* 1958)

Abstract—Maximum permissible levels of irradiation to the human body have been assigned from experience gained in the past with radium poisoning cases. Previous data on normal levels of radium concentration in the body are limited and show a wide variation. A comprehensive study of radium levels in the human skeleton has been made in this work using a larger number of samples.

The radium concentration in different bones within the skeleton has been measured and shows a range of less than ±15 per cent of the mean. In an urban population such as New York City the distribution of radium levels has been found to be log-normal. The average radium concentration from many localities scattered about the world is about 1×10^{-14} g Ra/g ash. The total range of all localities is from 0.3 to 3.0×10^{-14} g Ra/g ash.

These levels of radium-226 concentrations are approximately 1/3000 of the m.p.c. for this isotope.

INTRODUCTION

THE possibility that any ionizing radiation is potentially harmful requires accurate evaluation of all sources of radiation to the human body. At the present time this radiation dose is derived essentially from five sources: cosmic rays, γ-rays from the environment, potassium-40 in the body, fission products in the body particularly cesium-137 and strontium-90, and radium and its daughters. Of these sources, radium is important because of its central place in the assignment of the permissible levels of radiation that the human skeleton may receive.

Several investigators have reported measurements of the radium concentrations in the human body (Table 1) but the number of samples represented is small. All determinations were made by the emanation method on whole body ash except those of STEHNEY and LUCAS[3] which were accomplished by measuring exhaled radon.

The original data of KREBS[1] have been criticized by DUDLEY[6] on the grounds that his samples (0.1 g ash) were too small to permit accurate analyses and were particularly sensitive to contamination. Furthermore his results are very far removed from more recent and more refined measurements of MUTH and his co-workers.[5]

In the course of investigating the world-wide distribution of strontium-90 in the human skeleton, as part of Project Sunshine, a large and unique set of human bone samples has been assembled. It was felt that the opportunity existed for a more comprehensive study of radium levels in the world population and this was undertaken.

The specific objectives of the present work were (a) to determine the radium concentration of the various bones of the skeleton; (b) to examine the distribution of radium concentration in individuals from a given area, and (c) to evaluate the average and range of radium concentration of the human population.

EXPERIMENTAL PROCEDURE

About 20 g of single bone ash or total skeleton ash were dissolved in the minimum quantity of hot dilute hydrochloric acid. After the sample was cooled, it

* Lamont Geological Observatory Contribution No. 323.

† Present address: National Physical Laboratory, Teddington, Middlesex, England.

Table 1. *Radium contents of human bodies (previous investigations)*

Investigator	Origin	No. of samples	Mean (g Ra $\times 10^{14}$/g ash)	Method of measurement
KREBS[1]	Germany	18	500	Ra in whole body ash
HURSH and GATES[2]	New York State	21	5	Ra in whole body ash
STEHNEY and LUCAS[3]	Chicago	8	1.2	Rn exhalation
PALMER and QUEEN[4]	Pacific Northwest	50	1.6	Ra in whole body ash
MUTH *et al.*[5]	Germany	14	13	Ra in whole body ash

was filtered and any residue which remained was re-ashed in a porcelain crucible at 800°C for about 12 hr, or until there was no carbonaceous material visible. The ash remaining in the crucible was then dissolved in a few ml of 6 N HCl and added to the original solution, which was then diluted up to about 300 ml with distilled water. By this procedure the entire sample went into solution. The solution was transferred to a 500 ml round bottom flask which had been fitted with a ground-glass joint, a side-arm capillary tube, and a cold-finger condenser (Fig. 1). In all connections the minimum amount of grease was used. This assembly was connected to a vacuum system at the upper end of the condenser and nitrogen (which had been allowed to stand for at least 30 days) was bubbled through the solution for 5 min to flush out any residual radon which may have been present. The assembly was then closed off from the vacuum system, filled to a pressure of 1 atm with nitrogen and permitted to accumulate radon for at least 3 days. At the end of this period, the assembly was connected through an absorption column to a 4 l. stainless steel or copper ionization chamber. After the manifold and ionization chamber had been thoroughly evacuated, the flask assembly was opened to the system and the accumulated radon was flushed out of the solution into the chamber by bubbling nitrogen through the solution. When a pressure of 1 atm in the system was attained, the valve on the chamber was closed and the chamber disconnected from the system and transferred to the counting apparatus. The absorption column was 2 cm in diameter and about 20 cm long and was filled with "Ascarite" and "Drierite" to remove both water and HCl vapors. It was found necessary to change this absorption train after every fifteen runs.

The low-level radon counting apparatus, which consists of a vibrating reed electrometer used as a slow pulse device, an amplifier, discriminator, and a register, is described in detail elsewhere.[7]

The samples were normally counted overnight (16 hr) to obtain satisfactory counting statistics. A

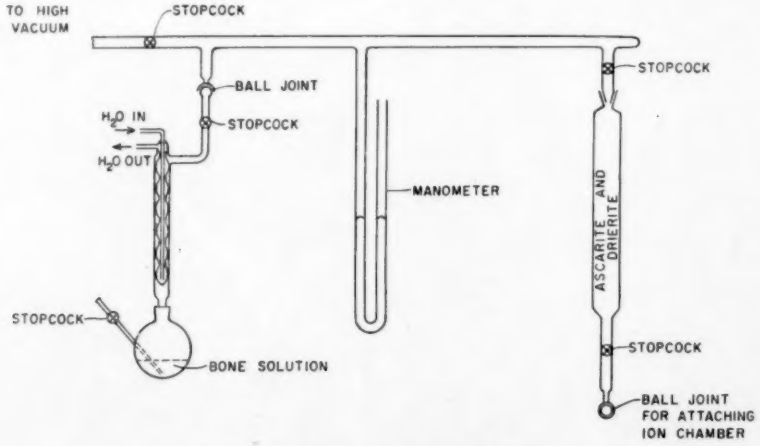


FIG. 1. Radon gas collection system.

Table 2. Reproducibility of secondary standards

Date (1958)	Std. chamber I (counts/hr)	Date (1958)	Std. chamber II (counts/hr)
15 Jan.	657 ± 20*	26 Jan.	625 ± 5*
16 Jan.	648 ± 13	27 Jan.	634 ± 15
17 Jan.	641 ± 11	28 Jan.	606 ± 10
20 Jan.	662 ± 10	3 Feb.	645 ± 10
21 Jan.	668 ± 12	5 Feb.	622 ± 12
22 Jan.	645 ± 15	7 Feb.	609 ± 13
24 Jan.	653 ± 13	13 Feb.	626 ± 6
26 Jan.	653 ± 5	17 Feb.	654 ± 6
27 Jan.	641 ± 15	18 Feb.	647 ± 10
28 Jan.	673 ± 13	19 Feb.	592 ± 13
3 Feb.	670 ± 10	21 Feb.	612 ± 10
5 Feb.	660 ± 12	2 Mar.	596 ± 15
	656 ± 11†		622 ± 20†

* Standard deviation on the count.
† Standard deviation on the average.

solution containing $(2.00 \pm 0.02) \times 10^{-12}$ g Ra, obtained from the National Bureau of Standards was measured about every 2 weeks and used as a primary standard to compute the efficiency of the counters. The average efficiency of this system is about 80 per cent. In between these calibrations the electronic stability of the counters was monitored every 2–3 days with two secondary standards. Each of these consists of a small quantity of radon maintained at a constant level of activity by a shielded piece of uraninite attached to the bottom of the ionization chamber. Table 2 illustrates the reproducibility of the secondary standards and it is evident that the electronic stability was excellent, generally being better than the counting statistics on an individual run. From the results of the background determinations shown in Table 3, it can be seen that these also fall essentially within that expected from counting statistics. The error over a restricted time interval is about ±1 count/hr. The background used in any particular sample calculation was the average background for the 2 weeks on either side of this sample and the error assumed for the background was the standard error on the mean. Since the sample count rates ranged from 20 counts/hr to 50 counts/hr an overnight count on a sample yielded an overall uncertainty of about 10 per cent at the 1 Σ level. Table 4 shows that repeat measurements at different times from the same bone solution are reproducible within this counting error.

Table 3. Reproducibility of background determinations

Date (1957)	Chamber I (counts/hr)	Date (1957)	Chamber II (counts/hr)
20 Aug.	8.3 ± 0.7*	17 Sept.	9.7 ± 0.8*
24 Aug.	7.0 ± 1.0	23 Sept.	9.1 ± 0.7
31 Aug.	12.6 ± 0.6	26 Sept.	8.8 ± 0.7
5 Sept.	8.6 ± 0.7	20 Sept.	8.6 ± 0.7
9 Sept.	9.5 ± 0.8	5 Sept.	11.1 ± 0.8
12 Sept.	9.6 ± 0.8	10 Oct.	8.5 ± 0.9
22 Sept.	8.0 ± 0.7	13 Oct.	9.2 ± 0.7
26 Sept.	5.7 ± 0.6	18 Oct.	11.5 ± 0.8
30 Sept.	8.6 ± 0.7	23 Oct.	12.0 ± 0.8
3 Oct.	9.9 ± 0.8	27 Oct.	7.7 ± 0.6
12 Oct.	6.9 ± 0.7	1 Nov.	9.0 ± 0.7
	8.8 ± 1.8†		9.6 ± 1.4†

* Standard deviation on the count.
† Standard deviation on the average.

Table 4. Typical reproducibility of sample determinations

Sample no.	Date (1957)	g Ra × 10^{14}/g ash
2119	20 Sept.	0.7 ± 0.1
	28 Sept.	0.9 ± 0.1
	11 Oct.	1.0 ± 0.1
	16 Oct.	0.8 ± 0.1
5294	16 Oct.	1.1 ± 0.1
	24 Oct.	1.3 ± 0.1
	27 Oct.	1.0 ± 0.1
5328	14 Oct.	0.6 ± 0.1
	20 Oct.	0.8 ± 0.2
	23 Oct.	0.7 ± 0.1
	27 Oct.	0.7 ± 0.1
2099	7 Sept.	1.0 ± 0.1
	12 Sept.	1.0 ± 0.1
	23 Sept.	1.4 ± 0.2

SAMPLING AND CONTAMINATION

The samples consisted of three types: (a) ash of single bones, (b) integrated ash of many single bones from the same locality, and (c) total skeleton ash from New York City cadavers. The cadavers had been used for anatomical instructional purposes and had therefore been

Table 5. *Radium contamination checks*
Experiment I *Sections of single femur ashed at different laboratories*

Wt. of ash (g)	Method of ashing	g Ra $\times 10^{14}$/g ash
32.5	N.Y.U. (stainless steel)	1.1 ± 0.1
24.0	Lamont (ceramic)	1.2 ± 0.1
27.1	N.Y.U.	0.9 ± 0.1
23.9	Lamont	0.9 ± 0.1
27.0	N.Y.U.	1.2 ± 0.2

Experiment II *Test of contamination from pulverizing (sections of single femur)*

Wt. of ash (g)	Method of ashing	g Ra $\times 10^{14}$/g ash
22.0	N.Y.U. (pulverized)	0.9 ± 0.1
26.0	Lamont (not pulverized)	0.8 ± 0.1
17.4	N.Y.U. (pulverized)	0.8 ± 0.1
26.2	Lamont (not pulverized)	0.8 ± 0.1

embalmed before being ashed. All of the ashed samples were finely ground before being processed according to the preceding description.

The possibility of contamination of the samples during ashing, pulverizing and chemical processing has been kept to a minimum by reducing the number of operations required to perform an analysis. Most of the samples were ashed at 800°C in clay pots in a muffle furnace at this laboratory but a number of the skeletons had to be ashed in stainless steel containers at some of the medical schools. The following experiments were performed to check each of these operations for possible contamination.

Five samples of bone from one femur shaft were subdivided into two groups, three samples being ashed at the New York University College of Medicine and the remaining two ashed at the Lamont Geochemical Laboratory. Each sample was then analyzed without being pulverized. The radium content is uniform within the experimental error indicating the absence of contamination arising from the different ashing procedures.

A second experiment was devised to determine whether the pulverizing process contributed any contamination. Four bone samples from a single femur shaft were again subdivided into two groups, two of them being ashed at the New York University College of Medicine and then pulverized at Lamont while the remaining two samples were just ashed at Lamont and then all four samples were analyzed for radium. The results from these experiments which are also shown in Table 5, prove that the pulverizing did not introduce any measurable contamination.

Results of blank determinations on the chemical procedure are shown in Table 6. These values are compared with the background counts of the ionization chamber employed. Since the values obtained are the same as the background determination, within the experimental error, no correction was applied for the blank.

Table 6. *Blank determinations*

Date (1957)	Can no.	Blank (counts/hr)	Background (counts/hr)
14 Sept.	4 B	9.7 ± 0.5	11.1 ± 1.6
20 Sept.	4 A	9.9 ± 0.5	9.3 ± 0.5
15 Oct.	2 A	17.6 ± 0.8	15.0 ± 1.0
15 Nov.	3 B	6.6 ± 0.5	4.7 ± 1.0

RESULTS AND DISCUSSION

(a) *Radium variation within the skeleton*

Previous work[8] on radium has established that it is deposited in bone but there is as yet no experimental data on the degree of variation of radium within the skeleton. Samples of different bones from each of eleven bodies have therefore been analyzed to determine this variation. All samples were analyzed at least in duplicate and the radium content is given in g Ra $\times 10^{14}$/g ash (Table 7). Each measurement has been corrected for background and the final value shown in the table is the mean of two or more determinations weighted according to the counting error of each measurement. The standard error on these results is about 10 per cent. The results show that there is a relatively large difference between the average radium

Table 7. Variation of radium within the skeleton (g Ra $\times 10^{14}$/g ash)

Bone	Body numbers											Normalized average
	1	2	3	4	5	6	7	8	9	10	11	
Vertebrae	1.2	0.9	0.8	1.7	0.8	1.0	1.0		2.3	1.8	0.9	1.2
Rib	1.5	1.2	0.6	1.1	0.8	0.9	0.8	1.1		2.4	0.8	1.1
Skull		1.2	0.5		0.5	1.1	0.5	0.7		2.4	1.0	0.9
Sternum									2.2	2.1	1.2	1.2
Pelvis					0.9	1.2	0.9	0.9	1.6	2.0	1.0	1.0
Femur	1.0		0.5	0.7	0.8	1.0	0.8	1.1	1.5	2.3	1.0	1.0
Fibula			0.5						1.5		1.2	1.0
Humerus	1.1	1.7	0.4	0.9	0.8	1.1	0.8	0.9	1.7	2.0	1.0	1.0
Radius	1.1				0.7		0.8					1.0
Tibia	1.0			1.0	0.6		0.6	1.1	1.5	2.2	1.2	1.0
Ulna	1.0	1.2			0.9		0.9	0.9	1.5		1.1	1.0
Body averages	1.1	1.2	0.6	1.1	0.8	1.0	0.8	1.0	1.7	2.2	1.0	

values of the eleven bodies. Within each body the variation for the majority of the samples is about ± 15 per cent of the average value for the skeleton radium content. In the final column in Table 7 is shown the average radium content for each bone type after normalizing the results to the average of the long bones = 1.0. It is clear that there is no systematic difference in radium content among the bones of the average skeleton, at least to within ± 15 per cent. Based on the work of SCHULERT *et al.*[9] with strontium-85 tracer experiments, and KULP[10] on environmental strontium-90, observable differences might be expected in adults if a person's diet in his last few years contained many times more radium than in his earlier life. The metabolically more active bones such as vertebrae and rib would show higher radium contents. Thus the strontium-90 data[10] reflect an uneven distribution of strontium-90 within the skeleton of adults, whereas in the case of common strontium, THURBER *et al.*[11] and HODGES *et al.*[12] have found a uniform distribution within the skeleton.

(b) *Radium abundance in cadavers from New York City*

To examine the distribution of radium concentrations in a single urban population group, total skeleton ash of many individuals from the New York City area was analyzed. The results for over one hundred and forty specimens are displayed in a histogram (Fig. 2) in which the number of individuals with a given radium concentration has been plotted against the natural logarithm of the radium burden of the individuals. Plotting the data in this manner results in a more symmetrical curve than if the radium concentration is used directly. This distribution law is common for trace elements in the geosphere as shown by AHRENS.[13] The average radium concentration of New York skeletons is therefore 0.8×10^{-14} g Ra/g ash with a standard deviation of $\pm$antiln 0.50. It seems unlikely that the distribution curve for any other major urban center would be greatly different from this area.

The majority of the individuals reported above died between the ages of fifty and eighty, with only a small number below forty. Upon plotting the radium concentration vs. age it was apparent that no trend with age exists. This finding is in agreement with the results of other workers[2–4] based on fewer samples. No difference was found, within the experimental error, for the average radium burden of twenty-five female samples when compared with the average for the remaining males.

(c) *World-wide variations in radium burdens*

The radium contents of human bone from a total of nineteen stations scattered over the

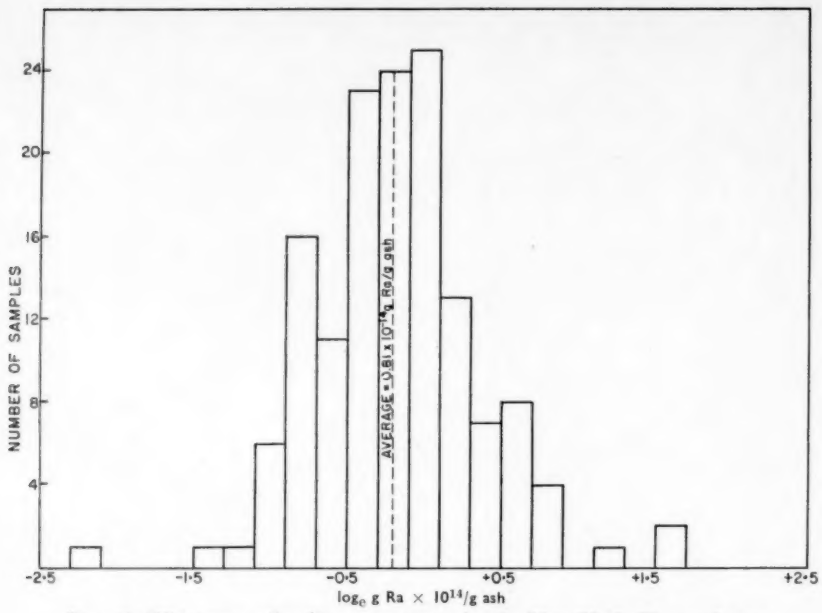


FIG. 2. Histogram of radium concentration in New York City cadavers.

world have been determined and the results are shown in Table 8. They are grouped in two sets, the earlier unpublished averages of single bone analyses obtained by KULP and ECKELMANN using similar techniques and the composite samples used in this work. The number of individuals sampled in each case is shown in the table.

The agreement was excellent for the Vancouver samples but our Chilean sample representing a composite of twenty-four bones was a factor of 10 lower than the average of only three samples shown in the earlier data. From Table 8 it is evident that the variation in radium contents from one city in the world to another is not great, the total range being about a factor of 10 ($0.3–3 \times 10^{-14}$ g Ra/g ash). If we accept the evaluations of DUDLEY[6] and TURNER *et al.*[14] that the main source of radium is in the food supply, it is probable that the small range in radium levels can be correlated with variations in the radium to calcium or radium to phosphorus ratios in soils. The geochemistry of the food producing regions for these areas requires further study, but, for example, the data suggest a higher Ra/Ca ratio in the area of the food source for Houston than in Vancouver if this theory is correct. It is interesting to note that the average radium concentration for the Boston area is about 1.4 higher than the New York average, but both regions have drinking water which is comparable in radium content (See HURSH[15]). This is further evidence in agreement with HURSH's conclusion that drinking water is not the single most important source of ingested radium, although locally[3] this may be the case.

(d) *Significance of radium as a radiation hazard*

From the analytical data presented in the previous sections, a world-wide arithmetical average for the radium content of man is about 1.2×10^{-14} g Ra/g ash. Assuming a standard adult skeleton contains about 3000 g bone ash, (or 1000 g calcium), 3.6×10^{-14} g Ra/g Ca and 0.36×10^{-10} g Ra per adult body are calculated. These figures are $\sim$1/3000 of the present maximum permissible concentration for this isotope.[16] It should be noted that these data are for bone alone without taking into account any radium which may be present in muscle and other tissue. MUTH *et al.*[5] have reported radium levels in soft tissue which would approximately double the total radium in the body.

Table 8

Location	No. of samples	g Ra $\times 10^{14}$/g ash
(A) *Average of single bone analyses*		
Canada (Vancouver)	7	0.6
Chile (Santiago)	3	3.0
Germany (Cologne and Bonn)	5	2.0
Switzerland (Zurich)	1	1.6
U.S.A. (Denver)	1	1.1
Venezuela (Caracas)	4	0.3
(B) *Composite samples—equal weights of individual ashed bones*		
Argentina (Cordoba)	18	1.1
Australia (Sydney)	22	1.0
Belgian Congo (Leopoldville)	14	2.2
Canada (Vancouver)	12	0.6
Chile (Santiago)	24	0.4
England (London)	21	1.5
Guatemala (Guatemala City)	29	0.5
Israel (Jerusalem)	13	3.6
India (Bombay)	12	0.8
Japan (Tokyo)	36	0.4
Puerto Rico (San Juan)	42	0.5
Union of South Africa (Durban)	13	1.1
U.S.A. (Boston)	77	1.4
U.S.A. (New York)	143	1.0
U.S.A. (Houston)	23	2.3

The dosage to the bone allowing for 25 per cent radon retention, 50 per cent excretion of RaD and an average skeleton of 7000 g, is therefore about 1 mrad/year for the average man. If the relative biological effectiveness of α-emitters is taken as 4 and that of β- and γ-emitters as 1 (MARINELLI[17]), then the skeletal radiation hazard from radium is about 3 mrems/year for the average man compared with about 150 mrems/year for the average annual background radiation.[18] It is clear therefore that the radium content of average man contributes about 2 per cent of his total damaging radiation.

If the distribution data are log-normal as was found in results from the New York City cadavers, then with a standard deviation such as was found in that example about 0.07 per cent of the population will have five times the average and $\sim$0.0002 per cent or two in one million will have ten times the average value.

It is also of interest to compare the radiation doses of radium and strontium-90 in the bone. The strontium-90 content in March 1958 for average man was 0.25 $\mu\mu$c/g Ca[10] and if the ratio of the energy release $Ra^{226}/Sr^{90} = 10$ and the r.b.e. for $\alpha/\beta = 4$ are assumed it can be seen that radium still presents a greater biological hazard to adults than strontium-90. Ra and Sr^{90} dosages will be comparable by 1978 when the average strontium-90 concentration is expected to be 0.9 $\mu\mu$c/g Ca provided there is no further bomb testing. In the case of young children in North America, the predicted strontium-90 content of their bones in 1978 is 1.5 $\mu\mu$c/g Ca[10] so that the effective radiation from strontium-90 will then be greater than the dosage from radium.

CONCLUSIONS

Based on the data which have been presented in this paper, the following conclusions are drawn:

(1) The radium variation within the skeleton of normal individuals is small. In the majority of cases the variation is not more than $\pm$15 per cent of the mean value.

(2) A distribution curve for the radium concentrations of New York City individuals has been determined from the analysis of over one hundred and forty cadavers. The curve is log-normal with an average value of 0.8 g Ra $\times$ 10^{14}/g ash and a standard deviation of $\sim\pm$15 per cent. This value is lower than previous measurements from other localities, the closest value being PALMER and QUEEN'S[4] result of 1.6×10^{-14} g Ra/g ash for the normal average in the Pacific Northwest.

(3) World-wide variations in radium burdens are not large and show a spread between $(0.3\text{–}3.0) \times 10^{-14}$ g Ra/g ash for different localities and a mean near 1×10^{-14} g Ra/g ash. Higher radium concentrations may possibly be found in regions where the Ra/Ca ratio in soils is high and in areas of abnormally high radium in drinking water.

(4) Radium concentrations in the bones of unexposed individuals are a factor of 3000 below the m.p.c. for this isotope. When compared with the total natural background radiation, radium contributes approximately 2 per cent of the dose.

The present strontium-90 dose in bones of adults is still much lower than the dose from radium. In the case of children the strontium-90 dosage will be greater than the radium dose by 1978.

Acknowledgments—This research was supported by the Division of Biology and Medicine, U.S. Atomic Energy Commission. The authors are indebted to Dr. A. R. SCHULERT of the Lamont staff, Dr. H. L. VOLCHOK of Isotopes, Incorporated and Dr. L. D. MARINELLI of the Argonne National Laboratory for helpful discussions. We wish to thank the following people who critically read the manuscript: Dr. R. A. DUDLEY, Dr. J. B. HURSH, Dr. W. F. LIBBY and Dr. H. Q. WOODARD. Miss ELIZABETH HODGES selected and prepared the bone ash samples.

REFERENCES

1. A. KREBS, *Strahlentherapie* **72,** 164 (1942).
2. J. B. HURSH and A. A. GATES, *Nucleonics* **7,** No. 1, 46 (1950).
3. A. F. STEHNEY and H. F. LUCAS JR., *International Conference on the Peaceful Uses of Atomic Energy*, A/Conf. 8/P/852 (1955).
4. R. F. PALMER and F. B. QUEEN, *Amer. J. Roent., Ra. Therapy and Nuclear Medicine* **79,** 521 (1958).
5. H. MUTH, A. SCHRAUB, K. AURAND and H. J. HANTKE, *Brit. J. Radiol.* Suppl. No. 7, 54 (1957).
6. R. A. DUDLEY, Private communication.
7. G. L. BATE, H. L. VOLCHOK and J. L. KULP, *Rev. Sci. Instrum.* **25,** No. 2, 153 (1954).
8. A. O. GETTLER and C. NORRIS. *J. Amer. Med. Ass.* **100,** 400 (1933).
9. A. R. SCHULERT, E. A. PEETS, D. LASZLO, H. SPENCER, M. CHARLES and J. SAMACHSOM, *J. Appl. Rad. Isotopes* **4,** 144 (1959).
10. J. L. KULP, *Bull. Schweiz. Akad. Med. Wissen.* In press.
11. D. L. THURBER, J. L. KULP, E. HODGES, P. W. GAST and J. M. WAMPLER, *Science* **128,** 256 (1958).
12. R. M. HODGES, N. S. MACDONALD, R. NUSBAUM, R. STEARNS, F. EZMIRLIAN, P. SPAIN and C. MCARTHUR, *J. Biol. Chem.* **185,** 519 (1950).
13. L. H. AHRENS, *Geochim. et Cosmochim. Acta* **5,** 49 (1954).
14. R. C. TURNER, J. M. RADLEY and W. V. MAYNEORD, *Nature, Lond.* **181,** 518 (1958).
15. J. B. HURSH, *Brit. J. Radiol.* Suppl. No. 7, 45 (1957).
16. *NBS Handbook* No. 52 (1953).
17. L. D. MARINELLI, Private communication.
18. L. R. SOLON, W. M. LOWDER, A. V. ZILA, H. D. LEVINE, H. BLATZ and M. EISENBUD, *Science* **127,** 1183 (1958).

Health Physics Pergamon Press 1959. Vol. 1, pp. 417–426. Printed in Northern Ireland

RADIATION AGING AND ITS RELATION TO THE PRINCIPLES OF HEALTH PHYSICS

H. P. YOCKEY
Health Physics Division, Oak Ridge National Laboratory
Oak Ridge, Tennessee

(*Received* 3 *September* 1958)

Abstract—The principles on which health physics is based are essentially those adopted in the 1920's. These principles have not been re-examined, although the advent of reactor and nuclear weapons technology has changed the scale of the radiation exposure problem by many orders of magnitude. This is the purpose of the present paper. Since radiation exposure is a problem of all human populations and since the consequences of environmental hazards are viewed in a different light by different cultures, it is necessary to make a clear distinction between questions of jurisprudence and questions of science. A review of some theories of radiation damage shows that a common feature is that the several aspects or manifestations can be subsumed under radiological aging. A number of authors have reached this conclusion from a pathological examination of tissues. There is now no well defined measure of what is meant by radiation damage; it is suggested that the amount of life shortening may serve as such a measure. Such a definition is scientifically correct and would be understood by executives, legislators and the public. The logical structure of the principles of health physics is greatly improved by this definition.

I. INTRODUCTION

WITHIN the few years since the invention of the nuclear reactor, radiological protection has grown from a health problem affecting a few industrial workers or a small number of physicians, dentists, physicists, radiochemists, cyclotron operators, X-ray machine operators, and other professional people to a problem which is the concern of every inhabitant of the globe. At the time the graphite reactor of this Laboratory—the first reactor to operate in the megawatt range of power—was put into operation, the world supply of Ra was somewhat less than 4 kc. But this reactor, alone, contained about 200 Mc of radioactivity. The possible hazard to health changed by the same order of magnitude or greater. Some of the more hazardous materials, Po^{210}, Sr^{90}, Pu^{239}, etc., which were little known and had been produced in minute quantities by cyclotrons, now appeared 1 lb or 100 lb amounts.

This new situation stimulated research in health physics, particularly radiation dosimetry, and also in genetics, radiology and other appropriate associated fields. New instrumentation has provided more accurate measurement of the physical quantities pertaining to the interaction of radiation and matter.

New physical and biological data have been integrated into the recommended rules and regulations of the International Commission on Radiological Units, the International Commission on Radiological Protection, and the National Committee on Radiation Protection. This has been done in a piecemeal fashion to improve and extend the same basic rules and regulations adopted by these organizations shortly after their formation in the late 1920's. No fundamental re-examination has been made of the principles on which these regulations are based.

In view of the greatly changed character of the problem and the greatly increased quantity of data, it seems justified to make such a re-examination. This paper will attempt (a) to show that ideas current in health physics, while perhaps acceptable before the reactor age, are no longer adequate; and (b) to suggest an approach to establishing new principles which take into account new experimental data and the social and economic aspects introduced by the reactor age.

II. THE DEVELOPMENT OF SOME SCIENTIFIC AND LEGAL ASPECTS OF THE PRINCIPLES OF RADIATION PROTECTION

(A) *The principles on which health physics is based and the need for an operational definition of radiation damage*

It was known in the 1920's as a result of the early work in radiobiology, radiology, the medical use of X-ray and natural radioisotopes and exposure to natural radioisotopes in industry, that ionizing radiation could produce serious health consequences, including death. It was, therefore, necessary that standards and units of some sort be agreed upon and put into practice. The rules and regulations recommended during this period were of a practical character and were not intended to possess a great fundamental sophistication or scientific validity. They proved to be so satisfactory for the times in which they were established, however, that they became enshrined by usage and custom and, notwithstanding some erosion or evolution, came to be thought of as being axiomatic.

Let us consider three of these rules. First, it was necessary to have some unit of measure of the radiation field. The roentgen[1] was defined and chosen as this unit. There has been considerable discussion and some lack of standardization of the use of this unit.[2] The rad was introduced in 1953 in order to clarify the distinction between exposure dose and absorbed dose.

The roentgen and the rad are very well established in radiology, health physics, radiobiology and radiation chemistry. They are quite useful for the purposes originally intended, which is the establishment of a scale factor for exposure dose and absorbed dose. It is a fact, often lost sight of, that they are not measures of radiation damage. For such a measure, reference must be made to the absorbing material and to the effect the radiation produces.

The actual measurement of the rad is almost always made in an ionization chamber, although the rule requires the measurement of the energy absorption in the irradiated material. This is a satisfactory method if the experiment is designed to satisfy the requirements of the Bragg–Gray principle and if the materials of the chamber are atomically similar to the medium of interest. Curiously enough, it is often stated that ionization chambers are used rather than devices to measure the total energy deposited per gram because the temperature rise due to the absorbed energy is too small to measure. PLATZMAN[3] has called attention to the fact that while this heat is related to the rad it may have little to do with damage to the material.

The physical concept which seems to be closest to the meaning of "damage," as used in ordinary discourse, is not absorbed energy—since energy is the ability to do work—but rather the *irreversible production of entropy*. Entropy is a measure of disorder in a material, and it is certainly the disordering effect of radiation that we are interested in from the point of view of measuring radiation damage. It is well to emphasize that the roentgen and the rad are intended to provide scale factors for radiation exposure and absorption; they are not direct measures of radiation damage.

The second principle adopted was a tolerance limit for *radiation exposure rate*. This was originally established at 0.2 r/day, but it has since been reduced considerably.[4,5] Furthermore, the idea of a tolerance limit has been extended to long term exposure as well because of the results of experiments on reduction of life span due to radiation exposure.

The third principle established a tolerance value for the body burden of Ra; later this too was changed to a maximum permissible value. These latter two principles were used after the invention of the nuclear reactor to develop maximum permissible exposure levels for internal and external radiation under many conditions and for many isotopes.

(B) *A distinction between scientific questions and legal matters; the desirability of their separation*

The adoption of these principles was well justified if for no other reasons than that the beneficial use of ionizing radiation and the progress of nuclear technology would have been prejudiced had there been any large scale or serious radiation accidents. New problems were dealt with by using, rather liberally, *ad hoc* notions or standards whenever such seemed to

be required. It does not seem to have been noticed that this trend has continued to such a degree that health physics has assumed a quite legalistic aspect.

Considerable confusion and contradiction in the health physics standards themselves has resulted. It is not possible to resolve these questions by applying proposed or established theories to experimental data. Much of this situation seems to be due to a lack of distinction between those matters capable of a scientific evaluation and those matters which lie properly in the field of jurisprudence.

It is a proper function of the scientist to define, measure and characterize radiation and the radiological damage it inflicts. It is not within his power to legislate. This is a function of the civil authorities in the country concerned or may be subject to international agreements between governments. These functions are different but, like a pair of gears, they must be meshed to avoid clashing. For example, it is currently believed on the basis of scientific evidence that any amount of radiation is harmful no matter how small. Does this mean that all radiation exposure should cease? Doubtless, this is socially and economically unacceptable. We do not live forever, regardless of the level of environmental hazards. What damage may then be regarded as acceptable? Clearly the answer does not lie in science but in jurisprudence and, therefore, will vary according to culture and circumstance.

At present, no account is taken of the fact that different societies have quite different attitudes toward all factors of human existence including life, health, sickness and death. These societies or cultural units, however they may be governed, are accustomed to making their own decisions with regard to environmental health hazards. As reactor technology grows in economic importance and in familiarity, they will do so in this field also where it has not already been done.

The scientist may play a role as an indispensable guide to the writing of appropriate legislation and he should take the initative in this direction. An interesting commentary on this subject has been given recently by STASON.[6]

III. SEVERAL APPROACHES WHICH LEAD TO PREMATURE AGING AS AN OPERATIONAL DEFINITION OF BIOLOGICAL RADIATION DAMAGE

(A) *The need for a single measure suitable for all kinds of radiation damage*

Usually when this subject is discussed a rather long list is given of the various manifestations of radiation damage; this is useful for some special purposes. But, generally, the contemplation of a swarm of different manifestations of a phenomenon confuses those common properties by which each one represents the phenomenon *per se*. It is apparent then that we are seeking a quantitative operational definition of the amount of radiation damage in general and not, at least primarily, a list of various kinds of radiation damage.

It seems to the present author that the amount of premature aging may serve as a measure of the total effect of all manifestations of radiation damage. Let us consider the arguments and the evidence in support of this definition of the amount of radiation damage. We wish to show here that premature aging as a measure of radiation damage is a feature of several current theories and also comes directly from pathological observations.

(B) *The concept of genetic noise and its role in radiation damage*

(1) *Radiation sensitivity of the genetic apparatus and the role of* DNA. It is the current view, suggested in 1929 by OTTO RAHN,[7] that the genetic structure of the cell is the part by far most sensitive to X-rays and other damaging radiation. LEA[8] was a strong advocate of this view, and he used it as the basis of his development of the target theory. The mechanism of this damage is believed to be gene mutation or chromosome aberration. The relation of these two mechanisms to the manifestation of radiation damage is a matter of debate.

GLASS[9] and HENSHAW[10] recently have given excellent reviews of the evidence for this view of the mechanism of biological radiation damage. If it is accepted that the genetic structure is the element with a controlling radiation sensitivity in the somatic cell and in the germ cell, then it

will be helpful to bring into our discussion some of the current ideas from biochemistry and the structure of macromolecules which bear on the question of the chemical constitution and function of the genetic material.

Considerable evidence has been obtained by a number of workers which is understood to show that the macromolecule which carries the genetic specificity is DNA. This has been reviewed recently by KUZIN.[11] WATSON and CRICK[12] have suggested a structure for this molecule. This structure turns out to be astonishingly featureless for a molecule which is associated with, or is the carrier of, the vast degree of genetical specificity in the living organism. The molecule resembles a rope ladder wound around a cylindrical pillar. The two helical strands are composed of ribose sugars and phosphate groups. The rungs of the ladder are formed of purine-pyrimidine pairs.

WATSON and CRICK explain three most puzzling biological problems on the basis of this model of DNA. First, they explain how this seemingly featureless high polymer is able to carry genetical specificity by suggesting that this specificity is carried by the exact order of purine–pyrimidine pairs which form the rungs of the rope ladder. The genetical specificity is thus actually written on DNA in a four-letter alphabet.

This ingenious idea fitted beautifully with certain data on protein structure and attempts by DOUNCE[13] to contrive a "template" for their formation. It has been shown that, indeed, the specificity of proteins is carried by the exact order of the amino acids in the polypeptide chain. Some rather astonishing changes in properties occur as a result of switching from one amino acid to another. In particular, it is believed that the substitution of a single amino acid in a polypeptide chain distinguishes sickle cell from normal hemoglobin.[14] Since all enzymes are proteins and since protein structure is known to be under genetic control, this suggestion of a relation between genetics and protein synthesis based on the properties of molecules was most illuminating.

Another specificity-bearing molecule is RNA. The structure of the compound is not known, but like DNA it is composed of four building blocks and presumably carries its specificity in a four-letter alphabet. RNA is believed to carry specificity from the DNA in the nucleus to bodies known as microsomes located in the cytoplasm. Proteins are believed to be formed in the microsomes. RNA is also the specificity-bearing molecule in some viruses, specifically in tobacco mosaic virus.

GAMOW[15] resolved a difficulty which seems to arise because proteins are formed of twenty amino acids, whereas DNA and RNA both presumably carry specificity in four-letter alphabets. He pointed out that this implied the existence of a code, say for example the four-symbol telegraph code (dot, dash, letter space, word space) which represents the twenty-six letter English alphabet plus numerals.

Second, WATSON and CRICK showed how their suggested DNA structure could explain replication. Suppose we imagine the rungs of rope ladder to be cut one-by-one and the two long helical strands to be unwound. The cutting is to be done at the hydrogen–hydrogen bonds which hold the purine to the pyrimidine. Now, two new rope ladders identical to the first may be formed given a source of the proper pieces, since guanidine may pair only with cytocine and thymine only with adenine. Other alternatives have been offered, but the general notion has been preserved that this model offers a molecular explanation for replication.

The third fundamental biological problem on which the molecular structure of DNA bears is a mechanism of mutation. According to the view that genetical specificity is carried by the exact order in which the purine–pyrimidine pairs are arranged, a mutation must be the result when there is a change in this order. WATSON and CRICK pointed out that rare tautomeric forms might exist which would allow a pairing other than that mentioned above. In particular, it is believed that adenine, in a rare tautomeric form, may pair with cytosine rather than thymine.

(2) *Genetic noise—establishing a common measure between aging and radiation damage.* It will be appreciated from the above discussion that the genetic apparatus functions much like a communication system. This is true with a

little broader generality than is usually understood. The body or somatic cells as well as the germ cells carry DNA, replicate and mutate. When speaking of the somatic cells, we shall use the term "somatic line" and for the germ cells "germ line". Thus a mutation in the somatic line may effect one cell or many cells in one individual. A mutation in the germ line may affect the progeny.

The genetic communication system must have noise—any real communication system does. This noise is the result of point mutations or other errors of duplication in the germ line which introduce alterations in the order of purine–pyrimidine pairs along the DNA molecule—one such mechanism is described above—or it may be the result of grosser disorganization in the gene code such as deletions, chromosome aberrations, etc. Any one of these kinds of disorder will be reflected by loss of information. We must, therefore, consider the biological effect of noise in the gene code. It should be understood, as a matter of fact, that the same kinds of genetic disorder are found to occur naturally as are caused by radiation.[16,17]

It is clear that if the level of genetic noise becomes too high the organism will die, because it will not be able to synthesize sufficient proteins with sufficient biochemical specificity. It is this fact which leads us directly to a mathematical theory of senescence, thermal killing and radiation damage. The source of noise makes little difference and one is, therefore, lead to a quantitative equivalence between the effect of these agents in the sense that the effect may be measured in the same terms.[18]

Classically, genetics has been concerned primarily with certain "markers" and their phenotypic manifestation, such as the color of the pea blossoms in GREGOR MENDEL's garden. Accordingly, we have been accustomed to acknowledge as genetic radiation damage only such identifiable markers as were found by MULLER[19] and by STADLER[20] to be induced by X-rays.

Since both the germ cells and the somatic cells carry DNA and accordingly, in that sense, have essentially the same genetic apparatus, the notion of genetic radiation damage may be broadened to include the somatic line as well as the germ line. In the light of this more extensive notion of the scope of genetics, and recognizing that the genetic apparatus operates as a communication system, we see an application of these ideas to a fourth and fifth field of biology. These are senescence and radiation damage.[18]

According to the argument given above one would expect aging to occur in the germ line both as a result of natural causes and as a result of radiation; this is indeed the case. LANSING[21] has reported the results of an experiment in two species of the rotifer, namely *Euchlanis triquetra* and *Philodina citrina*. These two species reproduce parthenogenically, and so the germ line resembles the somatic line as it does in many micro-organisms. By propagating only with the eggs laid near the end of the fertile period, LANSING was able to produce for each species a premature aging, which proceeded generation by generation and which ended in the death of the line. It was possible to reverse this trend by choosing, in some fertile generation, the eggs laid in the adolescent period.

RUSSELL[22] found a premature aging effect in the first generation progeny of male mice which had been exposed to neutrons from an atomic weapon. The dose levels were rather low, the highest being 186 reps for only two of the animals. Nevertheless, the existence of the effect seems to be established. These results seem to be related to the experiments of RUSS and SCOTT[23] in which male and female rats were exposed chronically to radium γ-radiation. In one series, both males and females received a total dose of 1480 r over a period of 168 days. The matings of each sex to an unirradiated fertile partner produced very few viable young, and those were short lived.

The experiments which show aging in the germ line are discussed in more detail elsewhere.[18] The point we wish to make here is that there seems to be a common measure between aging and radiation damage in the germ line as well as in the somatic line.

The current view of genetic damage places primary importance on the existence of specific mutations, dominant or recessive, which may be ascribed to radiation. The importance of this should not be gainsaid, but these lethal or

sub-lethal mutants are rare;[16] that is, the frequency is about 10^{-3} to 10^{-6} On the other hand, the premature aging affects all or nearly all the animals born of irradiated parents. As a matter of fact, those individuals affected by these rare mutations are shorter lived and less fertile than the normal population, and so the idea of premature aging as a measure of the damage induced by such mutation is not without application.

The premature aging due to radiation exposure seems to be a general effect as shown by the experimental evidence. Its magnitude reflects a weighted sum of many various and individually obscure effects which are deleterious to the general well-being of the animal and, therefore, affect its ability to survive. This is interpreted from information theory as being the result of genetic noise in the genetic information carried by DNA.

It was mentioned briefly above that the physical quantity which seems most closely related to damage in its commonly accepted meaning is the irreversible production of entropy rather than the absorption of energy. Genetic noise is measured by an entropy function and so it represents the irreversible production of genetic disorder and therefore resembles "damage" as ordinarily understood. But this has been discussed in more detail elsewhere.[18]

(C) BLAIR's *theory of radiation aging*

BLAIR[21] has given a theory of radiation damage which, although it fails in some details, presents a remarkably good description of the salient features. It postulates a single injury function which represents the sum of the reparable and the irreparable component of radiation damage. The magnitude of this injury function is measured by the average life shortening in the group of irradiated animals with respect to the average life-span of the control group. This theory deals with recovery by means of a simple first-order differential equation. The success of BLAIR's attempt to characterize a phenomenon, which in its details is quite complicated, is a principal contribution of his approach to the thesis of the present paper. That success shows that if one attempts to measure radiation damage in units of lifetime lost as compared to an unirradiated control, he can give a very good, if not perfect, account of the experimental results. BLAIR's injury function is related to the average amount of genetic noise introduced by the irradiation.[25]

An excellent analysis of BLAIR's theory and of some of the modifications made by others has been given by BERLIN and DIMAGGIO.[26]

(D) SACHER's *injury function derived from physiological variables*

SACHER[27] defines an injury function which is the weighted sum of several kinds of injury. The weight is assigned to each according to its importance for survival. In this theory, each such variable is characterized by a dose-sensitivity constant and a time response. The effect of the aging process is represented by adding a term to the radiation-induced damage. The death of the median animal occurs when this injury function reaches some lethal bound, specified by the nature of the experiment. With this formalism, SACHER was able to correlate data from aging and irradiation experiments over a very wide range of doses of several kinds of radiation to seven mammals.

SACHER has continued these studies and has recently shown that fluctuations in the physiological variables can account for an increased mortality rate in an aging population even though there is no initial distribution in sensitivity[28]. The logarithm of the probability of death per unit time is known as the Gompertz function. It is found to be linear for most wild-type animals which includes man. In SACHER's theory, the Gompertz function is regarded as an approximately linear measure of the physiological state.

There are some important parallels between SACHER's approach, information theory, and irreversible thermodynamics. The secular changes in the physiological variables are due, fundamentally, to irreversible changes in the permanent or self-reproducing macromolecules which govern the processes of biosynthesis in the organisms. The age changes are based on functions at the biochemical level and, therefore, have a meaning beyond the purely phenomenological.[29]

(E) JONES' *analysis of the radiation factor in longevity*

PEARL[30] showed that the life tables of man and *Drosophila* could be plotted as nearly identical curves if the time scale of the *Drosophila* population were adjusted so that 1 *Drosophila*-day equals about 1 man-year. The same thing is true for the mouse and the guinea pig, with appropriate adjustment of scale to suit the species. This indicates that whatever the aging process is, it affects the physiological function of many organisms in the same way.

LOEB and NORTHRUP[31] and BRODY[32] have suggested that this may be interpreted in terms of a physiological age. PEARL believed that the general form of the Gompertz function indicates that each individual has a certain amount of "vitality" and dies with its exhaustion. JONES[33,34,35] has given a detailed study of the effect of aging, disease, radiation damage and life expectancy in terms of the effect on the parameter of the Gompertz function.

JONES is able to show that diseases and death are not to be understood as misfortunes occurring at random and without which we would rival the Biblical patriarchs, but that every disease is to some extent related to every other disease. The tendency to disease and death in general may be correlated closely to a single quantity, which in this connection is called the physiological age.

For example, it can be shown[33] that the population of those Swedish males who develop cancer of the stomach is 5 years older when compared by means of the Gompertz function in 1951 than in 1911. Presumably the general improvement in health and the virtual elimination of some specific diseases has produced a population of individuals generally healthier and physiologically younger than that of 1911.

Radiation has essentially this same effect, but in reverse. The slope of the Gompertz function remains the same as the controls, but the death rate values reflect those of an older population. The notion that physiological aging is the biological result of radiation exposure may explain a number of very odd experimental facts. If the population is aged artificially, then the so-called late effects are simply manifestations of that aging. Specifically, the carcinogenic effects of radiation may be one of several manifestations of physiologic age in the same sense as mentioned above in connection with the general benefits of greater lifelong freedom from disease.

At any rate, JONES' analysis indicates that radiation damage, as well as other biological damage, may be defined and measured in terms of the life shortening relative to some standard control population. This is exactly what JONES[36] does for a number of measurable environmental and health factors. In the case of radiation exposure, he estimates that for man the life shortening is between 5 and 10 days per rad. RUSSELL's[22] result for mice is about 0.078 per cent/rad. Extrapolating this figure to man, one obtains a life shortening of about 20 days/rad with 5 and 35 days as the 95 per cent confidence limits. Because of the size of the statistical errors which must be associated with these estimates, the results of JONES and RUSSELL are not necessarily in disagreement.

FAILLA and MCCLEMENT[37] have analyzed the data of LORENZ *et al.*[38] on the life shortening of chronic irradiation of mice also by the use of the Gompertz function. They introduce a new feature, namely, the assignment of a fictitious or equivalent exposure rate to the normal aging process. The figure which they obtain is 12.8 r/day for mice. Since man lives twenty times longer than the mouse, this fictitious rate of aging is 0.64 r/day. This is, according to these authors, the dose rate which would result in a typical human-type life table for an organism otherwise immortal. Taking advantage of the additivity of normal and radiological aging, they arrive at a figure of 1 day/r for the rate of radiological aging. This value lies on the extreme range of JONES'[36] figures of one to 30 days/r and outside RUSSELL's extrapolation of his mouse data to man. These figures differ because of the techniques of dosimetry, because of different points of view taken by the several authors in the analysis of the data, and partly because of different values for the average life span of the mouse.

(F) *Premature aging and the concept of genetic load*

MULLER[39] has introduced the notion of the "load of mutations". The genetic make-up of

man is the result of the accumulation of beneficial and deleterious genes; the beneficial ones are preserved because they produce a successful population, the deleterious ones are eliminated at certain rates because eventually they cause the "genetic death" of some individual.

The frequency of mutations in man for specific undesirable traits has been obtained by NEEL, *et al.*[40] by a study of consanguineous matings in Japan. They obtained a value of about 2.8×10^{-5} to about 1.1×10^{-5} mutations per generation for specific traits. MORTON *et al.*[41] obtain an estimate between 1.5×10^{-5} and 0.6×10^{-5} per locus per generation for general deleterious mutation which, in the homozygous state, would kill an individual between birth and maturity. This estimate comes from an analysis of the longevity of progeny from consanguineous matings in the United States and France. MORTON *et al.* calculate a total mutation rate to such deleterious genes of from 0.03 to 0.05 per gamete per generation. According to these figures, the average individual carries from three to four such recessive lethal mutations.

Because it is a mutagenic agent and because only a few mutations are not harmful, radiation may be expected to increase the load of deleterious mutations in a population. At this writing, the effect of the genetic load on the entire life table has not been studied. The data are not extensive enough to do more than give total deaths at certain stages of maturity, namely, stillbirth, neonatal, infantile and juvenile. Nevertheless, it seems that a more complete approach to the problem would involve a study of the life shortening at all ages due to the genetic load. In that case, the suggestion we are entertaining would serve as a measure of the total effect of the increased genetic load relative to a control population resulting from any situation, including both radiation damage and in-breeding.

(G) *Pathological observations in support of the equivalence of radiation aging and normal aging*

RUSS and SCOTT[23] first observed clinical and pathological aspects in rats irradiated chronically with Ra γ-rays which they related to aging. This was confirmed by HENSHAW[42] in a more detailed and extensive study. He summarizes his results: "The gross pathologic changes were primarily those of aging: excessive graying of the hair, emaciation, marked loss of body fat, and generalized atrophy, the last-mentioned being most noticeable in the lymph nodes, spleen and testes." Similar observations and evaluations have been made by FURTH *et al.*[43] in mice exposed to radiation from a nuclear detonation and by UPTON *et al.*[44] in mice exposed to thermal neutrons.

IV. DISCUSSION

We have attempted to show that, for the purposes of health physics, radiation damage and normal aging may be regarded as equivalent. One arrives at this conclusion from a number of theoretical analyses as well as from the empirical observations of RUSS and SCOTT,[23] of HENSHAW,[42] and of later workers. The extent to which one may regard radiation damage and normal aging as being the same thing is discussed in recent reviews by ALEXANDER[45] and by UPTON.[46] Each of these reviewers concludes that there are very great similarities, but each is cautious about indicating identity. For the purposes of this article, one needs only the notion that, although radiation damage and normal aging may possibly have features which are not identical, nevertheless, they may be measured in the same units.

A strong case can be made for the thesis that the multitude of known and unknown, large and small, physiological and genetic effects of radiation may be subsumed under radiological aging. FAILLA[47] has pointed out that: "If the mechanism of aging brought about by radiation is essentially the same as that of the natural aging process, we can look to the future with much more confidence in setting up permissible limits of exposure".

Let us now again regard the health problem surrounding radiation exposure and examine them in the light of this argument. The roles of the roentgen, rad, rem and RBE are clarified. The first two of these concepts enable one to establish scale factors for exposure dose and absorbed dose; that is, the use of these quantities allows one to repeat experiments at different

times or to repeat those made in other laboratories. Of course, other details of such experiments must also be known. Neither of these units may be thought of as a unit of radiation damage.

The rem may be thought of from this point of view as a unit of damage in the sense of a weighted sum of all manifestations of radiation damage and is most conveniently measured in terms of life shortening. The rem, so defined, clearly has the property of additivity for various radiations. The RBE, by the same token, may be defined as the ratio of the absorbed dose of two different radiations to produce the same life shortening. Thus defined, both the rem and the RBE lose the vagueness with which they must be regarded by presently accepted notions[2] and become operationally definite quantities or relations.

How does this definition of damage fit in with problems of administration and jurisprudence? These problems are of two general kinds:

(a) *Problems of administration and legislation.* These are, the establishment of levels and other conditions of permissible occupational or environmental radiation exposure. Typical among the issues encountered here are the amount of shielding required, the design of remote handling equipment, or the amount of radioactivity that will be *permitted* to be discharged into the air or into surface water. In short, they involve decisions for action *before* the exposure. Furthermore, such decisions are made in general terms to cover a class of cases.

(b) *Problems of jurisprudence.* These questions involve decisions which must be made after an exposure. They are made only after an exhaustive examination of the details of a particular case. Has the plaintiff been damaged; if so, how much? What liability has the defendant? What compensation, if any, is due the plaintiff, his progeny or his estate?

Radiation accidents which result in death in 30 days can hardly lead to more than the usual questions concerning the relation of the death to the accident and the legal liabilities involved. The situation is different when there is an exposure from which the plaintiff recovers and exhibits "no observable clinical symptoms or effects". From the point of view of the present paper, damage has been sustained in a quantitative amount by the plaintiff and his progeny which are conceived after the exposure, if any. It is up to the particular system of jurisprudence to deal with the case as seems most suitable. But this is done continually in dealing with human affairs and, indeed, it is one of the *raisons d'etre* of any system of jurisprudence.

At any rate, given the definition of radiation damage suggested here, the scientist would assume his proper role as an expert witness, and the results of his testimony would be readily understood by administrators, legislators, courts and the general public.

The fact that there is no definition of radiation damage currently accepted regrettably results in the use of phrases to which each writer or reader attaches his own individual meaning and which are, therefore, meaningless. Typical among these are: (a certain radiation exposure) "need cause no alarm to the individual on his own account"; "no undue concern"; "possible biological hazards"; "negligible damage"; and so forth. Such remarks as these and also "concentrations (of radioactive isotopes) that it is safe to accumulate" furthermore imply a threshold. There is little experimental evidence support that idea. Even the term "hazard" has its illusory aspects since it implies that it is possible to be exposed to radiation and escape the consequences. In terms of the concept that all types of radiation damage are subsumed under radiation aging, the situation is divested of such operationally meaningless and misleading phraseology.

It is to be hoped that the greatly changed character of the physical, biological, and medical problems surrounding the social and economic aspects of the reactor age will stimulate a re-examination of the fundamental concepts. This will lead to the establishment of permissible exposure levels at values consistent with the benefits to be derived from the circumstances surrounding the exposure rather than to the mistaken impression that they are "safe" or of "no undue concern". The formulation of concepts appropriate to deal with health physics problems is urgently needed. It is hoped this article may help to call attention to this fact.

REFERENCES

1. L. S. Taylor, *Health Physics* **1,** 306 (1958).
2. *Report of the International Commission on Radiological Units and Measurements* (*ICRU*) 1956, National Bureau of Standards Handbook 62 (1957); S. N. Ardashnikov and N. S. Chetverikov, *Soviet J. Atomic Energy* **3,** 1097 (1957).
3. R. L. Platzman, *Phil. Mag.* **44,** 497 (1953).
4. L. S. Taylor, *Health Physics* **1,** 3 (1958).
5. L. S. Taylor, *Health Physics* **1,** 97, 200 (1958).
6. E. Blythe Stason, *Legal Problems of Liability and Financial Protection Connected with Radiation Injuries* (A/CONF.15/P/2352). Paper presented at Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September 1958.
7. O. Rahn, *J. Gen. Physiol.* **13,** 179 (1929); *J. Gen. Physiol.* **14,** 315 (1930).
8. D. E. Lea, *Actions of Radiations on Living Cells* The University Press, Cambridge, (1946).
9. B. Glass, *Science* **126,** 241 (1957).
10. P. S. Henshaw, *Radiology* **69,** 30 (1957).
11. A. M. Kuzin, *Biological Effect of Ionizing Radiation in the Light of Contemporary Opinions on the Nature of D.N.A.* pp. 273–284. Izvestia Akad. Nauk SSSR (1957).
12. J. D. Watson and F. H. C. Crick, *Nature, Lond.* **171,** 964 (1953); *Cold Springs Harbor Symposia Quant. Biol.* **18,** 123 (1953); *Nature, Lond.* **171,** 737 (1953).
13. A. L. Dounce, *Enzymologia* **15,** 251 (1952).
14. V. M. Ingram, *Nature, Lond.* **178,** 792 (1956); *Proceedings of the Conference on Hemoglobin*, National Research Council, Washington, D.C. (1957).
15. G. Gamow, *Kgl. Danske Videnskab. Selskab. Biol. Medd.* **22,** 1 (1954).
16. *The Biological Effects of Atomic Radiation*, Summary Reports, National Academy of Sciences—National Research Council Washington (1956); J. V. Neel, *Proc. Natl. Acad. Sci.* **43,** 736 (1957).
17. *The Hazards to Man of Nuclear and Allied Radiations*, Medical Research Council, Great Britain (1956).
18. H. P. Yockey, A study of aging, thermal killing, and radiation damage by information theory. In *Symposium On Information Theory In Biology*. (Edited by H. P. Yockey, R. L. Platzman and H. Quastler). Pergamon Press, London (1958).
19. H. J. Muller *Science* **66,** 84 (1927).
20. L. J. Stadler, *Science* **68,** 186 (1928); *Proc. Natl. Acad. Sci.* **14,** 69 (1928).
21. A. I. Lansing *Proc. Natl. Acad. Sci.* **34,** 304 (1948).
22. W. L. Russell *Proc. Natl. Acad. Sci.* **43,** 324 (1957).
23. S. Russ and G. M. Scott *Brit. J. Radiol.* **10,** 619 (1937); *Brit. J. Radiol.* **12,** 440 (1939).
24. H. A. Blair, A quantitative description of latent injury from ionizing radiation. In *Symposium On Information Theory In Biology*. (Edited by H. P. Yockey, R. L. Platzman and H. Quastler). Pergamon Press, London (1958).
25. H. P. Yockey, *Rad. Res.* **5,** 146 (1956).
26. N. I. Berlin and F. L. Dimaggio, Armed Forces Special Weapons Project, Washington Report No. 608.
27. G. A. Sacher, *J. Natl. Cancer Inst.* **15,** 1125 (1955). Reparable and irreparable injury: A survey of the position in experiment and theory. In *Radiation Biology and Medicine* (Edited by W. D. Claus). Addison Wesley (1958).
28. G. A. Sacher, *Radiology* **67,** 250 (1956).
29. G. A. Sacher, Entropic contributions to mortality and aging. In *Symposium On Information Theory In Biology* (Edited by H. P. Yockey, R. L. Platzman and H. Quastler). Pergamon Press, London (1958).
30. R. Pearl, *The Rate of Living*. Knopf, New York (1928).
31. J. Loeb and J. H. Northrup, *J. Biol. Chem.* **32,** 103 (1917).
32. S. Brody, *J. Gen. Physiol.* **6,** 245 (1923).
33. H. B. Jones, *Trans. N. Y. Acad. Sci.* **18,** 298 (1956).
34. H. B. Jones, *Kaiser Found. Med. Bull.* **4,** 329 (1956).
35. H. B. Jones, Aging disease and life expectancy. In *Advances in Biological and Medical Physics* (Edited by J. H. Lawerence and C. A. Tobias) Vol. IV. (1956).
36. H. B. Jones, *Proc. Health Phys. Soc. First Annual Meeting* p. 114 (1956).
37. G. Failla and P. McClement *Amer. J. Roent. Rad. Ther. Nuc. Med.* **78,** 946 (1957).
38. E. Lorentz, *et al.*, *Biological Effects of External X- and Gamma-Radiation*. National Nuclear Energy Series IV, 22-B. McGraw-Hill, New York (1954).
39. H. J. Muller, *Amer. J. Human Genet.* **2,** 111 (1950) General survey of mutational effects of radiation. In *Radiation Biology and Medicine* (Edited by W. D. Claus). Addison Wesley (1958).
40. J. V. Neel, M. Kodani, R. Brewer and R. C. Anderson, *Amer. J. Human Genet.* **1,** 156 (1949).
41. N. E. Morton, J. F. Crow and H. J. Muller, *Proc. Natl. Acad. Sci.* **42,** 855 (1956).
42. P. S. Henshaw, *J. Natl. Cancer Inst.* **4,** 513 (1944).
43. J. Furth, A. C. Upton, K. W. Christernbery, W. H. Benedict and J. Moshman, *Radiology* **63,** 562 (1954).
44. A. C. Upton, J. Furth and K. W. Christenberry, *Cancer Res.* **14,** 682 (1954).
45. P. Alexander, *Gerontologia* **1,** 174 (1957).
46. A. C. Upton, *J. Geront.* **12,** 306 (1957).
47. G. Failla, *Radiology* **69,** 23 (1957).

Health Physics Pergamon Press 1959. Vol. 1, pp. 427–441. Printed in Northern Ireland

U^{235} FISSION PRODUCT DECAY SPECTRA AT VARIOUS TIMES AFTER FISSION*

ANN T. NELMS and J. W. COOPER
Nuclear Physics Section National Bureau of Standards, Washington, D.C.

(*Received* 12 *September* 1958)

Abstract—Tabulations are given of the β- and γ-ray spectra produced by the total products of U^{235} fission at times after fission from 31.2 min to 119 years. Spectra are also given which exclude radiation from fission products that form volatile compounds.

1. INTRODUCTION

THE products of U^{235} fission which are present after long times consist of several hundred nuclides ranging in mass number from 70 to 161. These products are formed by two processes, namely (a) fission and (b) chain decay of the initial fission fragments. Once fission has taken place in a given manner, the known chain decay schemes and half-lives of the various products serve as a basis for computing the activities of further products at later times. Just what the initial fission fragments are will depend upon how fission occurs.

Fission product mass distributions may be estimated by observing fissions where the products are relatively long-lived. The determination of the charge distribution is somewhat more difficult. Several theories of the most probable charge distribution in fission have been advanced and are discussed in reference (1b).

The most complete information available on the time dependent activity of product nuclides for U^{235} fission is given by BOLLES and BALLOU[2] who have computed activities as a function of time for the average population of nuclides produced by a standard number (10^4) of U^{235} fissions. Two tabulations are given, one using the charge distribution theory of GLENDENIN *et al.*[1b] the other that of PRESENT.[3] The results given in this paper are based upon these tabulations.

2. γ-RAY SPECTRA

(a) *Method of calculation*

The energy distribution of γ-radiation for any time after fission depends not only on the activity of the fission products but also upon the energy and photon abundance of each γ-emitting disintegration. MILLER[4] has tabulated this information for all of the known γ-emitting nuclides listed by BOLLES and BALLOU.

By combining the information contained in references (2) and (4) we obtain photon energy distributions for various times after fission. Energy distributions are represented by dividing the range between 0 and 5.899 MeV into twenty-two "boxes". For purposes of tabulation each energy "box" is represented by energy E_l. "Box" limits (ε_l) are assumed to be midway between these representative energies on a logarithmic scale. Representative energies, "box" limits and "box" widths are given in Table 1.

The calculation procedure is straightforward. Let $L_j(t_i)$ be the activity in disintegrations/sec for the jth nuclide at time t_i after fission and $N_j(E_r)$ the abundance of photons from this nuclide expressed in photons/disintegration with energy E_r. Then the number of photons/sec emitted by the jth nuclide at time t_i with energies between ε_l and ε_{l+1} is:

$$\sum_{\varepsilon_l < E_r < \varepsilon_{l+1}} N_j(E_r) L_j(t_i)$$

* Work supported by Federal Civil Defense Agency.

Table 1. *Energy "boxes" used for γ-spectra*

Energy (E_l) (MeV)	"Box" limits (ε_l)	"Box" size ($\varepsilon_{l+1} - \varepsilon_l$)
	0.0	
0.0340		0.0381
	0.0381	
0.0425		0.0110
	0.0491	
0.0567		0.0152
	0.0643	
0.0729		0.0220
	0.0863	
0.1021		0.0279
	0.1142	
0.1277		0.0333
	0.1475	
0.1703		0.0429
	0.1904	
0.2128		0.0427
	0.2331	
0.2554		0.0525
	0.2856	
0.3193		0.0831
	0.3687	
0.4257		0.0976
	0.4663	
0.5108		0.1048
	0.5711	
0.6386		0.1663
	0.7374	
0.8514		0.1953
	0.9327	
1.0217		0.2096
	1.1424	
1.2772		0.3324
	1.4748	
1.7029		0.3907
	1.8655	
2.0435		0.4193
	2.2848	
2.5545		0.6648
	2.9496	
3.4059		0.7814
	3.7310	
4.0871		0.8385
	4.5695	
5.1089		1.3298
	5.8993	

An estimate of the spectral density $K(t_i, E_l)$ at time t_i and energy E_l is obtained by summing over all fission product nuclides and dividing by the energy box width as indicated by the formula:

$$K(t_i, E_l) = \sum_j \sum_{\varepsilon_l < E_r < \varepsilon_{l+1}} N_j(E_r) L_j(t_i) / (\varepsilon_{l+1} - \varepsilon_l) \quad (1)$$

(b) *Results*

Tables 2–5 give four tabulations of $K(t_i, E_l)$. Tables 2 and 3 have been calculated after the yield theories of GLENDENIN *et al.*[1b] and PRESENT,[3] respectively. These tabulations include the nuclides listed in Table 6. Tables 4 and 5 have been calculated in the same way as Tables 2 and 3 but omitting the elements most likely to form volatile compounds or remain in the gaseous state, namely Br, Kr, Ru, I and Xe. For each of the above tabulations the total product activity $T(t_i) = \sum_j L_j(t_i)$ has also been calculated for the species included. These total product activities are given for each of the four tabulations along with the total product activities for all species as given by BOLLES and BALLOU in Table 7.

(c) *Comparison with experiment and other results*

Calculations similar to those given here have been reported previously.[5,6] These papers predict the fission spectra produced by reactors which is a function of both the period of operation and the time after shutdown. Accordingly, they compare with the results of the present report only for cases where the operation time is small compared to the time after shutdown.

Experimental spectra have been obtained from operating reactors and from photofission of U^{235}.[7,8] In both cases comparisons can be made with this tabulation only when the time of irradiation is small compared to the time after fission considered.

Fig. 1 shows a comparison between data from reference (7) and from Table 2. The ordinate scale is only applicable to the calculated data from GLENDENIN. The experimental curve was superimposed arbitrarily to compare shapes. Agreement is quite good except in the neighborhood of 3.5 MeV where the experimental

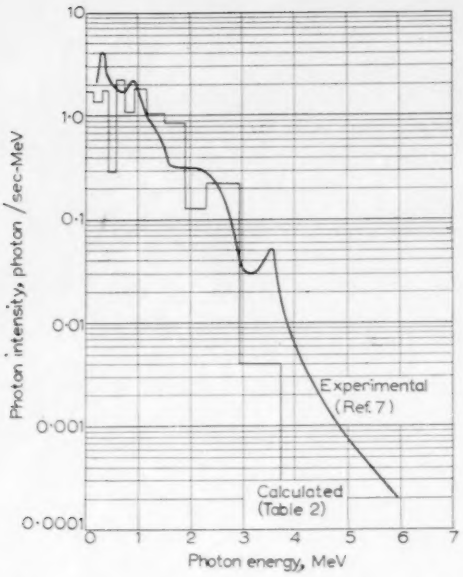


FIG. 1. Comparison of the calculated gamma spectra from the yield theory of GLENDENIN ($t = 31.2$ min) and the experimental γ-spectra ($t = 25.8$ min). Note: The scale shown is for the calculated spectra. The absolute intensity of the experimental results could not be determined.

results seem to show a line not listed by MILLER. No satisfactory explanation of this discrepancy has been found to date.

Fig. 2 gives the total γ-activity (photons/sec) summed over all energies as a function of time after fission for the data of Tables 2 and 4. The general shape of these curves was found to agree very well with the experimental results of BORST[1a] for short irradiations, especially in the range from 1 to 500 days.

3. β-RAY SPECTRA

(a) *Method of calculation*

β-particles which are ejected from atomic nuclei during the process of spontaneous transformation have total energies ranging from zero to a maximum energy τ_n characteristic of the specific mode of disintegration n. The statistical distribution of these energies $S_{nj}(\tau)$ is the β-spectrum of the nth transition of nuclide j. A nuclide may disintegrate in one or more fashions each having a probability of occurrence known as the branching ratio. The sum of the distributions for all modes of disintegration in nuclide j is the β-spectrum of the material.[9] In this paper spectra are calculated from the

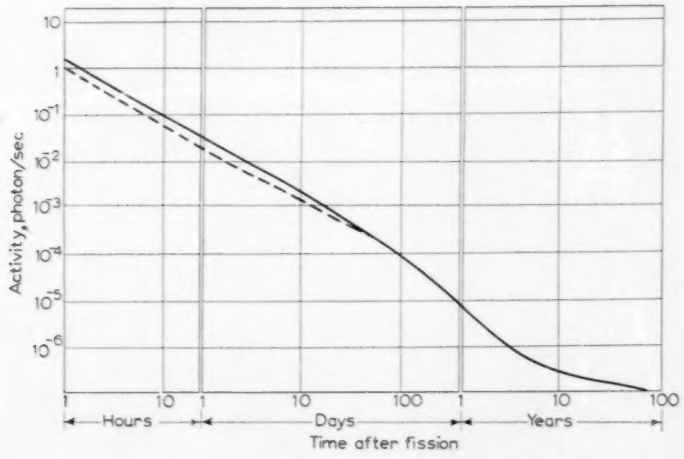


FIG. 2. Total γ-activity versus time after fission:
from the yield theory of GLENDENIN.
——— total products
– – – without volatile materials
(Br, Kr, Ru, I and Xe)

Table 2. Fission spectra for 10^4 U^{235} *fissions in photons/sec* (MeV) (yield theory of GLENDENIN *et al.*)

Energy (MeV)	31.2 min	1.12 hr	2.40 hr	5.16 hr ($\times 10^{-1}$)	11.1 hr ($\times 10^{-1}$)	23.8 hr ($\times 10^{-1}$)	2.13 days ($\times 10^{-2}$)	4.57 days ($\times 10^{-2}$)	9.82 days ($\times 10^{-2}$)	21.1 days ($\times 10^{-3}$)
0.0340	3.0995	1.6526	0.7479	2.9063	0.9635	0.3076	2.0128	1.4970	0.7958	2.6434
0.0425	3.4983	1.8511	0.9936	4.5186	2.2657	1.6564	10.0765	4.0328	1.4331	7.6538
0.0567	0.0669	0.0369	0.0291	0.2553	0.2124	0.1608	0.9169	0.2791	0.0299	0.0715
0.0729	0.0311	0.0053	0.0022	0.0339	0.0564	0.0903	1.2019	1.1175	0.6026	1.4204
0.1021	0.5117	0.2353	0.1094	0.3865	0.0704	0.0355	0.2955	0.2261	0.1542	0.8135
0.1277	1.6130	0.9368	0.2386	0.3992	0.2077	0 1475	1.1071	0.7307	0.4260	2.3756
0.1703	1.8716	0.3865	0.1580	0.6705	0.1902	0.0457	0.2096	0.1045	0.0374	0.1104
0.2128	3.2662	0.7527	0.3089	2.0312	0.9466	0.2642	1.4018	0.8025	0.2541	0.2262
0.2554	0.2150	0.1342	0.0991	1.0007	1.0443	0.7262	1.8631	0.3404	0.1398	0.6165
0.3193	2.4445	1.1559	0.3826	1.8683	0.6111	0.1946	1.0976	0.5129	0.2177	0.8940
0.4257	1.7863	1.1440	0.3128	0.3825	0.0288	0.0142	0.1030	0.0656	0.0399	0.1986
0.5108	0.2862	0.2603	0.2117	1.6943	1.0829	0.4789	1.7948	0.5729	0.2512	1.3246
0.6386	2.3486	0.9399	0.3758	1.4871	0.6107	0.3114	1.1910	0.3608	0.1119	0.3004
0.8514	1.0419	0.8776	0.5241	2.1553	0.9547	0.3828	1.1697	0.3700	0.1563	0.6938
1.0217	1.7652	0.6981	0.2863	0.8873	0.3104	0.0877	0.2658	0.0989	0.0203	0.0132
1.2772	1.0063	0.2415	0.0230	0.0934	0.0504	0.0166	0.0435	0.0168	0.0040	0.0032
1.7029	0.8536	0.4405	0.1219	0.2522	0.0920	0.0228	0.0919	0.0809	0.0643	0.3404
2.0435	0.1253	0.0870	0.0446	0.2019	0.0770	0.0164	0.0210	0.0075	0.0024	0.0033
2.5545	0.2453	0.1202	0.0358	0.1083	0.0383	0.0077	0.0061	0.0025	0.0023	0.0129
3.4059	0.0040	0.0017	0.0004	0.0009	0.0002	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	0.0002	—	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-3}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-7}$)
0.0340	0.8181	2.7665	0.4200	0.3165	0.1389	0.8702	0.6569	4.8682	2.7919	0.8724
0.0425	3.1577	8.9682	2.1486	8.2451	2.4456	2.2014	0.4177	3.6209	2.7310	1.4944
0.0567	0.0521	0.3536	0.2154	1.1861	0.3734	0.3994	0.0179	0.0015	—	—
0.0729	0.0834	0.1841	0.1406	0.8143	0.2485	0.2230	0.0061	0.0005	—	—
0.1021	0.2338	0.4263	0.1873	0.9940	0.3024	0.3588	0.1236	1.0602	0.7994	0.4374
0.1277	1.1134	3.2582	0.5128	1.5404	0.4509	0.3310	0.0012	—	—	—
0.1703	0.0261	0.0154	0.0003	0.0023	0.0016	0.0075	0.0015	0.0004	—	—
0.2128	0.0014	0.0016	0.0015	0.0147	0.0145	0.1409	0.1316	1.1579	0.8733	0.4778
0.2554	0.1685	0.1896	0.0375	0.0377	—	—	—	—	—	—
0.3193	0.1694	0.0716	0.0010	0.0088	0.0087	0.0844	0.0790	0.6957	0.5247	0.2871
0.4257	0.0533	0.0381	0.0019	0.0070	0.0047	0.0225	0.0046	0.0012	—	—
0.5108	0.5039	1.2216	0.1712	0.1575	0.0496	0.0658	0.0044	0.0135	0.0015	—
0.6386	0.1483	0.8224	0.2754	0.5442	0.2236	1.6092	1.3802	10.6188	6.0986	1.9058
0.8514	0.4052	2.6171	1.0260	0.9823	0.0057	0.0012	—	—	—	—
1.0217	0.0002	0.0013	0.0008	0.0050	0.0018	0.0023	—	—	—	—
1.2772	0.0002	0.0014	0.0004	0.0006	0.0001	0.0001	—	—	—	—
1.7029	0.0916	0.0565	0.0020	0.0119	0.0036	0.0040	0.0009	0.0058	0.0020	0.0002
2.0435	0.0008	0.0060	0.0044	0.0257	0.0075	0.0060	—	—	—	—
2.5545	0.0035	0.0021	—	0.0002	—	—	—	—	—	—
3.4059	—	—	—	0.0002	0.0002	0.0020	0.0013	0.0051	0.0006	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 3. Fission spectra for 10^4 U^{235} *fissions in photons/sec* (MeV) (yield theory of PRESENT)

Energy (MeV)	31.2 min	1.12 hr	2.40 hr	5.16 hr ($\times 10^{-1}$)	11.1 hr ($\times 10^{-1}$)	23.8 hr ($\times 10^{-1}$)	2.13 days ($\times 10^{-2}$)	4.57 days ($\times 10^{-2}$)	9.82 days ($\times 10^{-2}$)	21.1 days ($\times 10^{-3}$)
0.0340	2.7803	1.4023	0.6488	2.7656	0.9671	0.3164	2.0643	1.5102	0.7962	2.6439
0.0425	3.6360	1.8651	0.9938	4.5176	2.2654	1.6564	10.0765	4.0328	1.4331	7.6538
0.0567	0.0699	0.0372	0.0291	0.2553	0.2124	0.1608	0.9169	0.2791	0.0299	0.0715
0.0729	0.0341	0.0057	0.0022	0.0337	0.0563	0.0902	1.2019	1.1178	0.6029	1.4214
0.1021	0.5306	0.2325	0.1098	0.3895	0.0729	0.0374	0.3058	0.2288	0.1542	0.8132
0.1277	1.5888	0.8638	0.2273	0.4378	0.2364	0.1676	1.2156	0.7587	0.4273	2.3756
0.1703	1.8932	0.3822	0.1569	0.6654	0.1898	0.0465	0.2180	0.1068	0.0375	0.1104
0.2128	3.3441	0.7572	0.3067	2.0200	0.9379	0.2596	1.3636	0.7812	0.2477	0.2198
0.2554	0.2194	0.1311	0.1037	1.0263	1.0450	0.7224	1.8458	0.3454	0.1403	0.6176
0.3193	2.4701	1.1578	0.3826	1.8603	0.6101	0.1955	1.1075	0.5183	0.2207	0.9035
0.4257	1.7255	1.1076	0.2968	0.3591	0.0320	0.0170	0.1181	0.0695	0.0401	0.1986
0.5108	0.3065	0.2660	0.2118	1.6888	1.0829	0.4764	1.7917	0.5701	0.2511	1.3238
0.6386	2.3254	0.9274	0.3776	1.5083	0.6213	0.3149	1.2039	0.3563	0.1122	0.2993
0.8514	0.9781	0.8122	0.4868	2.0948	0.9659	0.3874	1.1862	0.3692	0.1564	0.6924
1.0217	1.7596	0.6536	0.2611	0.8377	0.3115	0.0889	0.2719	0.0996	0.0204	0.0129
1.2772	1.0203	0.2430	0.0228	0.0914	0.0489	0.0166	0.0432	0.0166	0.0040	0.0031
1.7029	0.8171	0.4197	0.1165	0.2472	0.0919	0.0228	0.0917	0.0806	0.0643	0.3403
2.0435	0.1263	0.0862	0.0438	0.1972	0.0747	0.0164	0.0205	0.0073	0.0024	0.0032
2.5545	0.2449	0.1196	0.0355	0.1062	0.0369	0.0077	0.0060	0.0025	0.0023	0.0129
3.4059	0.0040	0.0017	0.0004	0.0009	0.0001	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-3}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-7}$)
0.0340	0.8181	2.7656	0.4199	0.3165	0.1389	0.8702	0.6569	4.8682	2.7919	0.8724
0.0425	3.1577	8.9682	2.1486	8.2451	2.4456	2.2014	0.4177	3.6209	2.7310	1.4944
0.0567	0.0521	0.3536	0.2154	1.1861	0.3734	0.3994	0.0179	0.0015	—	—
0.0729	0.0836	0.1841	0.1406	0.8143	0.2485	0.2230	0.0061	0.0005	—	—
0.1021	0.2333	0.4251	0.1871	0.9940	0.3024	0.3588	0.1236	1.0602	0.7994	0.4374
0.1277	1.1134	3.2582	0.5128	1.5404	0.4509	0.3310	0.0012	—	—	—
0.1703	0.0261	0.0154	0.0003	0.0023	0.0016	0.0075	0.0015	0.0004	—	—
0.2128	0.0014	0.0076	0.0015	0.0147	0.0145	0.1409	0.1316	1.1579	0.8733	0.4778
0.2554	0.1688	0.1897	0.0375	0.0377	—	—	—	—	—	—
0.3193	0.1716	0.0717	0.0010	0.0088	0.0087	0.0844	0.0790	0.6957	0.5247	0.2871
0.4257	0.0533	0.0381	0.0019	0.0070	0.0047	0.0225	0.0046	0.0012	—	—
0.5108	0.5039	1.2217	0.1712	0.1576	0.0496	0.0658	0.0044	0.0135	0.0015	—
0.6386	0.1485	0.8224	0.2754	0.5442	0.2236	1.6092	1.3802	10.6188	6.0986	1.9058
0.8514	0.4052	2.6171	1.0260	0.9823	0.0057	0.0012	—	—	—	—
1.0217	0.0002	0.0013	0.0008	0.0050	0.0018	0.0023	—	—	—	—
1.2772	0.0002	0.0014	0.0004	0.0006	0.0001	0.0001	—	—	—	—
1.7029	0.0916	0.0565	0.0020	0.0119	0.0036	0.0040	0.0090	0.0058	0.0020	0.0002
2.0435	0.0008	0.0060	0.0044	0.0257	0.0075	0.0060	—	—	—	—
2.5545	0.0035	0.0021	—	0.0002	—	—	—	—	—	—
3.4059	—	—	—	0.0002	0.0002	0.0020	0.0013	0.0051	0.0006	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 4. The data of Table 2 with volatile constituents removed

Energy (MeV)	31.2 min	1.12 hr	2.40 hr	5.16 hr ($\times 10^{-1}$)	11.1 hr ($\times 10^{-1}$)	23.8 hr ($\times 10^{-1}$)	2.13 days ($\times 10^{-2}$)	4.57 days ($\times 10^{-2}$)	9.82 days ($\times 10^{-2}$)	21.1 days ($\times 10^{-3}$)
0.0340	2.7161	1.3063	0.4695	1.2108	0.3754	0.1472	0.8073	0.4723	0.2515	1.3480
0.0425	3.4983	1.8511	0.9936	4.5186	2.2657	1.6564	10.0765	4.0328	1.4331	7.6538
0.0567	0.0614	0.0316	0.0250	0.2362	0.2086	0.1602	0.9121	0.2745	0.0255	0.0370
0.0729	0.0308	0.0049	0.0011	0.0104	0.0091	0.0068	0.0379	0.0121	0.0031	0.0220
0.1021	0.5117	0.2354	0.1094	0.3865	0.0704	0.0355	0.2955	0.2261	0.1542	0.8135
0.1277	1.6130	0.9368	0.2386	0.3992	0.2077	0.1475	1.1071	0.7307	0.4260	2.3756
0.1703	1.8233	0.3440	0.1261	0.4953	0.1392	0.0421	0.2091	0.1045	0.0374	0.1097
0.2128	3.1243	0.6300	0.2201	1.5869	0.8449	0.2627	1.3893	0.7940	0.2521	0.2255
0.2554	0.1975	0.1050	0.0502	0.2344	0.1076	0.0800	0.5660	0.2979	0.1246	0.5584
0.3193	2.3974	1.1128	0.3474	1.6435	0.5148	0.1609	0.8600	0.3159	0.0899	0.4053
0.4257	1.5253	0.9543	0.2171	0.1642	0.0203	0.0142	0.1026	0.0652	0.0396	0.1957
0.5108	0.2132	0.1872	0.1356	0.9952	0.5573	0.1680	0.4738	0.2401	0.1651	0.8926
0.6386	2.3198	0.9127	0.3518	1.3007	0.4917	0.2420	0.7490	0.0996	0.0226	0.1722
0.8514	0.6782	0.5152	0.2912	1.6052	0.8805	0.3377	0.8437	0.1863	0.0997	0.6425
1.0217	1.4878	0.4477	0.1386	0.5952	0.2967	0.0792	0.1997	0.0590	0.0077	0.0018
1.2772	0.9908	0.2271	0.0109	0.0068	0.0058	0.0041	0.0223	0.0076	0.0011	0.0005
1.7029	0.5650	0.3671	0.1139	0.2235	0.0817	0.0187	0.0650	0.0673	0.0603	0.3368
2.0435	0.0889	0.0543	0.0205	0.0679	0.0183	0.0015	0.0013	0.0002	0.0001	0.0012
2.5545	0.2102	0.0914	0.0164	0.0097	0.0021	0.0002	0.0019	0.0025	0.0023	0.0129
3.4059	0.0040	0.0017	0.0004	0.0009	0.0002	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-3}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-7}$)
0.0340	0.7336	2.6907	0.4102	0.3148	0.1389	0.8702	0.6569	4.8682	2.7919	0.8724
0.0425	3.1577	8.9682	2.1486	8.2451	2.4456	2.2014	0.4177	3.6209	2.7310	1.4944
0.0567	0.0296	0.2632	0.2025	1.1839	0.3734	0.3994	0.0179	0.0015	—	—
0.0729	0.0206	0.1833	0.1406	0.8143	0.2485	0.2230	0.0061	0.0005	—	—
0.1021	0.2338	0.4263	0.1873	0.9940	0.3024	0.3588	0.1236	1.0602	0.7994	0.4374
0.1277	1.1134	3.2582	0.5128	1.5404	0.4509	0.3310	0.0012	—	—	—
0.1703	0.0258	0.0152	0.0003	0.0023	0.0016	0.0075	0.0015	0.0004	—	—
0.2128	0.0014	0.0016	0.0015	0.0147	0.0145	0.1409	0.1316	1.1579	0.8733	0.4778
0.2554	0.1613	0.1888	0.0375	0.0377	—	—	—	—	—	—
0.3193	0.1089	0.0644	0.0010	0.0088	0.0087	0.0844	0.0790	0.6957	0.5247	0.2871
0.4257	0.0514	0.0304	0.0008	0.0069	0.0047	0.0225	0.0046	0.0012	—	—
0.5108	0.2407	0.1635	0.0204	0.1305	0.0489	0.0598	0.0007	—	—	—
0.6386	0.1310	0.7722	0.2683	0.5430	0.2236	1.6092	1.3802	10.6188	6.0986	1.9058
0.8514	0.4052	2.6171	1.0260	0.9823	0.0057	0.0012	—	—	—	—
1.0217	0.0002	0.0013	0.0008	0.0050	0.0018	0.0023	—	—	—	—
1.2772	0.0002	0.0014	0.0004	0.0006	0.0001	0.0001	—	—	—	—
1.7029	0.0916	0.0565	0.0020	0.0119	0.0036	0.0040	0.0009	0.0058	0.0020	0.0002
2.0435	0.0008	0.0060	0.0044	0.0257	0.0075	0.0060	—	—	—	—
2.5545	0.0035	0.0021	—	0.0002	—	—	—	—	—	—
3.4059	—	—	—	0.0002	0.0002	0.0020	0.0013	0.0051	0.0006	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 5. The data of Table 3 with volatile constituents removed

Energy (MeV)	31.2 min	1.12 hr	2.40 hr	5.16 hr ($\times 10^{-1}$)	11.1 hr ($\times 10^{-1}$)	23.8 hr ($\times 10^{-1}$)	2.13 days ($\times 10^{-2}$)	4.57 days ($\times 10^{-2}$)	9.82 days ($\times 10^{-2}$)	21.1 days ($\times 10^{-3}$)
0.0340	2.3941	1.0603	0.3744	1.0865	0.3871	0.1573	0.8619	0.4851	0.2514	1.3470
0.0425	3.6360	1.8651	0.9938	4.5176	2.2654	1.6564	10.0765	4.0328	1.4331	7.6538
0.0567	0.0644	0.0318	0.0250	0.2362	0.2086	0.1602	0.9121	0.2745	0.0255	0.0370
0.0729	0.0339	0.0052	0.0011	0.0104	0.0091	0.0068	0.0379	0.0121	0.0031	0.0220
0.1021	0.5306	0.2325	0.1098	0.3895	0.0729	0.0374	0.3058	0.2288	0.1542	0.8132
0.1277	1.5888	0.8638	0.2273	0.4378	0.2364	0.1676	1.2156	0.7587	0.4273	2.3756
0.1703	1.8454	0.3405	0.1255	0.4923	0.1400	0.0428	0.2175	0.1067	0.0375	0.1097
0.2128	3.2039	0.6368	0.2197	1.5816	0.8397	0.2581	1.3511	0.7727	0.2458	0.2192
0.2554	0.1967	0.0969	0.0507	0.2393	0.1120	0.0832	0.5835	0.3025	0.1248	0.5584
0.3193	2.4234	1.1153	0.3480	1.6393	0.5157	0.1623	0.8699	0.3185	0.0900	0.4053
0.4257	1.4662	0.9188	0.2011	0.1409	0.0236	0.0169	0.1177	0.0691	0.0397	0.1957
0.5108	0.2126	0.1866	0.1351	0.9918	0.5567	0.1676	0.4741	0.2399	0.1651	0.8926
0.6386	2.2917	0.8961	0.3508	1.3103	0.5012	0.2469	0.7687	0.1013	0.0227	0.1722
0.8514	0.5832	0.4468	0.2638	1.5759	0.8923	0.3433	0.8649	0.1902	0.0998	0.6425
1.0217	1.4630	0.4026	0.1211	0.5704	0.2982	0.0806	0.2068	0.0608	0.0078	0.0018
1.2772	1.0049	0.2288	0.0109	0.0068	0.0058	0.0041	0.0225	0.0077	0.0011	0.0005
1.7029	0.5316	0.3487	0.1085	0.2183	0.0818	0.0187	0.0652	0.0673	0.0603	0.3368
2.0435	0.0896	0.0540	0.0203	0.0671	0.0181	0.0015	0.0013	0.0003	0.0001	0.0012
2.5545	0.2102	0.0912	0.0164	0.0096	0.0020	0.0002	0.0019	0.0025	0.0023	0.0129
3.4059	0.0040	0.0017	0.0004	0.0009	0.0001	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	0.0002	—	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-3}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-7}$)
0.0340	0.7332	2.6898	0.4101	0.3148	0.1389	0.8702	0.6569	4.8682	2.7919	0.8724
0.0425	3.1577	8.9682	2.1486	8.2451	2.4456	2.2014	0.4177	3.6209	2.7310	1.4944
0.0567	0.0296	0.2632	0.2025	1.1839	0.3734	0.3994	0.0179	0.0015	—	—
0.0729	0.0206	0.1833	0.1406	0.8143	0.2485	0.2230	0.0061	0.0005	—	—
0.1021	0.2333	0.4251	0.1871	0.9940	0.3024	0.3588	0.1236	1.0602	0.7994	0.4374
0.1277	1.1134	3.2582	0.5128	1.5404	0.4509	0.3310	0.0012	—	—	—
0.1703	0.0258	0.0152	0.0003	0.0023	0.0016	0.0075	0.0015	0.0004	—	—
0.2128	0.0014	0.0016	0.0015	0.0147	0.0145	0.1409	0.1316	1.1579	0.8733	0.4778
0.2554	0.1613	0.1888	0.0375	0.0377	—	—	—	—	—	—
0.3193	0.1089	0.0644	0.0010	0.0088	0.0087	0.0844	0.0790	0.6957	0.5247	0.2871
0.4257	0.0514	0.0304	0.0008	0.0069	0.0047	0.0225	0.0046	0.0012	—	—
0.5108	0.2407	0.1636	0.0204	0.1305	0.0489	0.0598	0.0007	—	—	—
0.6386	0.1310	0.7722	0.2683	0.5430	0.2236	1.6092	1.3802	10.6188	6.0986	1.9058
0.8514	0.4052	2.6171	1.0260	0.9823	0.0057	0.0012	—	—	—	—
1.0217	0.0002	0.0013	0.0008	0.0050	0.0018	0.0023	—	—	—	—
1.2772	0.0002	0.0014	0.0004	0.0006	0.0001	0.0001	—	—	—	—
1.7029	0.0916	0.0565	0.0020	0.0119	0.0036	0.0040	0.0009	0.0058	0.0020	0.0002
2.0435	0.0008	0.0060	0.0044	0.0257	0.0075	0.0060	—	—	—	—
2.5545	0.0035	0.0021	—	0.0002	—	—	—	—	—	—
3.4059	—	—	—	0.0002	0.0002	0.0020	0.0013	0.0015	0.0006	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 6. γ-Emitting nuclides

Nuclide	Masses	Nuclide	Masses
Zn 30	72, 74	In 49	117, 118, 119
Ga 31	72, 73, 74	In_1 49	115
Ge 32	75, 78	Sn 50	125, 127
Ge_2 32	77	Sb 51	125, 126, 127, 128, 129, 131
As 33	77, 78	Te 52	132, 134
Se_1 34	81, 83	Te_1 52	125, 127, 129, 131, 133
Br 35	83, 84	Te_2 52	129, 131, 133
Kr 36	83, 87, 88	I 53	131, 132, 133, 134, 135
Kr_1 36	85	Xe 54	138
Kr_2 36	85	Xe_1 54	131, 133, 135
Rb 37	88, 89, 91	Xe_2 54	133, 135
Sr 38	91, 92, 93	Cs 55	138, 139
Y 39	90, 92, 93, 94	Ba 56	137, 139, 140, 141, 142
Y_1 39	91	La 57	140, 141, 142, 143
Y_2 39	91	Ce 58	141, 143, 144, 145, 146
Zr 40	95, 97	Pr 59	144, 145, 146
Nb 41	98	Nd 60	147, 149, 151
Nb_1 41	95, 97	Pm 61	149, 151, 152, 153
Nb_2 41	95, 97	Sm 62	151, 153, 155, 156, 158
Mo 42	99, 101	Eu 63	155, 156, 157, 158
Tc 43	101, 102	Gd 64	159
Ru 44	103, 105	Tb 65	161
Rh 45	103, 106, 107		
Rh_1 45	105		
Rh_2 45	105		
Pd 46	111, 112		
Ag 47	109, 111, 112		
Cd 48	117, 120		
Cd_1 48	115		
Cd_2 48	115		

Table 7. Total fission product activities for various times after 10^4 U^{235} fissions in disintegrations/sec

Time after fission	*B* and *B* (GLENDENIN)	Table 2	Table 4	*B* and *B* (PRESENT)	Table 3	Table 5
31.2 min	2.336	2.252	1.974	2.307	2.224	1.939
1.12 hr	1.071	1.066	0.888	1.039	1.030	0.8528
2.40 hr	0.4352	0.4352	0.329	0.4200	0.4197	0.3160
5.16 hr	0.1846	0.1845	0.136	0.1822	0.1821	0.1349
11.1 hr	0.0816	0.0815	0.0596	0.0819	0.0817	0.0601
23.8 hr	0.0327	0.0327	0.0228	0.0329	0.0329	0.0231
2.13 days	0.0122	0.0119	(2)836	0.0122	0.0120	(2)847
4.57 days	(2)478	(2)446	(2)299	(2)479	(2)446	(2)300
9.82 days	(2)222	(2)196	(2)131	(2)222	(2)196	(2)131
21.1 days	(2)103	(3)849	(3)663	(2)103	(3)849	(3)663
45.3 days	(3)446	(3)371	(3)330	(3)447	(3)371	(3)330
97.3 days	(3)184	(3)160	(3)148	(3)184	(3)160	(3)148
208.0 days	(4)614	(4)542	(4)524	(4)615	(4)542	(4)524
1.20 years	(4)167	(4)138	(4)137	(4)167	(4)138	(4)137
2.60 years	(5)605	(5)396	(5)388	(5)605	(5)396	(5)388
5.58 years	(5)248	(5)107	(5)101	(5)248	(5)107	(5)101
12.0 years	(5)151	(6)679	(6)640	(5)151	(6)679	(6)640
25.7 years	(6)908	(6)451	(6)437	(6)908	(6)451	(6)437
55.3 years	(6)409	(6)205	(6)203	(6)409	(6)205	(6)203
119.0 years	(7)954	(7)491	(7)491	(7)954	(7)491	(7)491

The number in parentheses indicates the number of zeros immediately to the right of the decimal point.

Table 8. β-Emitting nuclides

Nuclide	Masses	Nuclide	Masses
Zn 30	72, 74	Cd_2 48	115
Ga 31	72, 73, 74	In 49	117, 118, 119
Ge 32	75, 77, 78	In_1 49	115
As 33	77, 78, 79	Sn 50	121, 123, 125, 126, 127
Se 34	81, 83	Sb 51	125, 126, 127, 128, 129 131
Br 35	83, 84	Te 52	127, 129, 132, 134
Kr 36	87, 88	Te_1 52	131
Kr_1 36	85	Te_2 52	131, 133
Kr_2 36	85	I 53	131, 132, 133, 134, 135
Rb 37	88, 89, 91	Xe 54	133, 138
Sr 38	89, 90, 91, 92, 93	Xe_1 54	135
Y 39	90, 92, 93, 94	Xe_2 54	135
Y_2 39	91	Cs 55	137, 138, 139
Zr 40	95, 97	Ba 56	139, 140, 141, 142
Nb_2 41	95, 97	La 57	140, 141, 142, 143
Nb 41	98	Ce 58	141, 143, 144, 145, 146
Mo 42	99, 101, 102	Pr 59	143, 144, 145, 146
Tc 43	101, 102	Nd 60	147, 149, 151
Ru 44	103, 105, 106	Pm 61	147, 149, 150, 151, 152, 153
Rh_2 45	105	Sm 62	151, 153, 155, 156, 158
Rh 45	106, 107		
Pd 46	109, 111, 112	Eu 63	155, 156, 157, 158
Ag 47	111, 112, 113, 115	Gd 64	159
Cd 48	118, 120	Tb 65	161
Cd_1 48	115		

Table 9. Fission spectra for 10^4 U^{235} *fissions in electrons/sec* (MeV) (yield theory of GLENDENIN *et al.*)

Energy (MeV)	31.2 min ($\times 10^{-1}$)	1.12 hr ($\times 10^{-1}$)	2.40 hr ($\times 10^{-1}$)	5.16 hr ($\times 10^{-2}$)	11.1 hr ($\times 10^{-2}$)	23.8 hr ($\times 10^{-2}$)	2.13 days ($\times 10^{-3}$)	4.57 days ($\times 10^{-3}$)	9.82 days ($\times 10^{-3}$)	21.1 days ($\times 10^{-3}$)
0.0340	5.6709	2.9030	1.1814	4.8200	2.4955	1.3540	8.0099	4.8839	2.4856	1.1040
0.0425	6.1940	3.1601	1.2896	5.2954	2.7289	1.4662	8.5819	5.2066	2.6478	1.1685
0.0567	6.8764	3.4875	1.4304	5.9393	3.0368	1.6043	9.2151	5.5409	2.8161	1.2307
0.0729	7.4462	3.7492	1.5473	6.5118	3.2999	1.7086	9.5859	5.6980	2.8955	1.2522
0.1021	8.0782	4.0094	1.6749	7.2386	3.6110	1.8002	9.6276	5.5839	2.8437	1.2125
0.1277	8.3352	4.0796	1.7246	7.6456	3.7654	1.8144	9.2503	5.2242	2.6737	1.1356
0.1703	8.3650	3.9896	1.7243	8.0068	3.8738	1.7679	8.2427	4.3903	2.2758	0.9895
0.2128	8.1301	3.7741	1.6702	8.1378	3.9028	1.7138	7.3591	3.6399	1.8932	0.8554
0.2554	7.8714	3.5700	1.6093	8.1531	3.9105	1.6902	6.8717	3.1475	1.6007	0.7405
0.3193	7.8907	3.5364	1.5806	8.0023	3.8176	1.6256	6.1805	2.5152	1.2170	0.5768
0.4257	7.8722	3.5566	1.5394	7.5085	3.5610	1.4988	5.4225	2.0249	0.9269	0.4264
0.5108	7.5359	3.4653	1.4737	6.9837	3.2909	1.3710	4.8384	1.7176	0.7586	0.3483
0.6386	6.7511	3.2059	1.3313	5.9900	2.7653	1.1370	3.9614	1.3190	0.5496	0.2530
0.8514	5.3932	2.6233	1.0286	4.0817	1.7555	0.6994	2.3917	0.7031	0.2586	0.1199
1.0217	4.4775	2.2013	0.8572	3.1619	1.2458	0.4594	1.4202	0.3593	0.1312	0.0663
1.2772	3.6259	1.7211	0.6406	2.2257	0.7878	0.2395	0.5762	0.1092	0.0418	0.0231
1.7029	2.6298	1.1413	0.3938	1.3324	0.4605	0.1228	0.2659	0.0339	0.0089	0.0050
2.0435	1.7929	0.7036	0.2270	0.7205	0.2255	0.0476	0.0930	0.0104	0.0028	0.0025
2.5545	0.9548	0.3341	0.1032	0.2967	0.0658	0.0059	0.0018	0.0008	0.0008	0.0008
3.4059	0.3349	0.1021	0.0319	0.1085	0.0195	0.0008	—	—	—	—
4.0871	0.1261	0.0346	0.0096	0.0438	0.0099	0.0004	—	—	—	—
5.1089	0.0037	0.0011	0.0003	0.0016	0.0003	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-4}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-8}$)
0.0340	6.1976	3.8129	1.5202	3.4875	1.0617	3.7626	1.5102	6.1310	2.8841	9.2033
0.0425	6.4222	3.8784	1.5302	3.4918	1.0695	3.8718	1.5671	6.2943	2.8562	8.3862
0.0567	6.5147	3.8003	1.4758	3.3972	1.0610	3.9241	1.6086	6.4384	2.7745	7.0333
0.0729	6.3177	3.5095	1.3327	3.1271	1.0065	3.8354	1.6223	6.6807	2.8081	6.4104
0.1021	5.5436	2.7280	0.9762	2.4471	0.8492	3.4399	1.5905	7.2579	3.1031	7.0239
0.1277	4.7435	2.0209	0.6642	1.8207	0.6845	2.9289	1.4895	7.4837	3.2457	7.3002
0.1703	3.7776	1.3421	0.3880	1.1096	0.4390	2.0379	1.2614	7.4432	3.2904	7.3164
0.2128	3.1778	1.0372	0.2799	0.6961	0.2695	1.4195	1.0688	7.0530	3.1415	6.8926
0.2554	2.7117	0.8060	0.1960	0.4243	0.1876	1.2328	0.9718	6.4405	2.8429	6.1356
0.3193	2.0435	0.5203	0.1155	0.2498	0.1319	0.9787	0.7781	5.1029	2.2058	4.5963
0.4257	1.4394	0.3402	0.0840	0.1924	0.0854	0.5215	0.3974	2.5212	1.0163	1.8517
0.5108	1.2273	0.3151	0.0821	0.1713	0.0590	0.2620	0.1851	1.1318	0.4157	0.5944
0.6386	0.9694	0.2901	0.0810	0.1734	0.0557	0.2092	0.1446	0.9127	0.3451	0.4944
0.8514	0.5640	0.2295	0.0725	0.1817	0.0605	0.2182	0.1488	0.9325	0.3431	0.4413
1.0217	0.3589	0.1698	0.0600	0.1817	0.0622	0.2123	0.1420	0.8841	0.3182	0.3737
1.2772	0.1378	0.0772	0.0374	0.1685	0.0600	0.1861	0.1203	0.7470	0.2662	0.2986
1.7029	0.0358	0.0288	0.0221	0.1314	0.0448	0.1035	0.0584	0.3625	0.1292	0.1450
2.0435	0.0228	0.0200	0.0153	0.0896	0.0279	0.0359	0.0111	0.0685	0.0244	0.0274
2.5545	0.0076	0.0067	0.0052	0.0303	0.0092	0.0080	—	—	—	—
3.4059	—	—	—	0.0001	—	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 10. *Fission spectra for* 10^4 U^{235} *fissions in electrons/sec* (MeV) (yield theory of PRESENT)

Energy (MeV)	31.2 min ($\times 10^{-1}$)	1.12 hr ($\times 10^{-1}$)	2.40 hr ($\times 10^{-1}$)	5.16 hr ($\times 10^{-2}$)	11.1 hr ($\times 10^{-2}$)	23.8 hr ($\times 10^{-2}$)	2.13 days ($\times 10^{-3}$)	4.57 days ($\times 10^{-3}$)	9.82 days ($\times 10^{-3}$)	21.1 days ($\times 10^{-3}$)
0.0340	5.3434	2.6632	1.1081	4.7272	2.4792	1.3507	7.9651	4.8539	2.4773	1.1038
0.0425	5.8466	2.9032	1.2110	5.1958	2.7117	1.4631	8.5372	5.1761	2.6395	1.1683
0.0567	6.5107	3.2122	1.3457	5.8320	3.0192	1.6019	9.1734	5.5114	2.8081	1.2306
0.0729	7.0764	3.4643	1.4590	6.4001	3.2826	1.7072	9.5506	5.6714	2.8885	1.2523
0.1021	7.7331	3.7283	1.5865	7.1265	3.5959	1.8012	9.6088	5.5657	2.8394	1.2129
0.1277	8.0351	3.8175	1.6406	7.5388	3.7532	1.8176	9.2481	5.2148	2.6721	1.1363
0.1703	8.1667	3.7784	1.6533	7.9151	3.8673	1.7743	8.2651	4.3943	2.2783	0.9906
0.2128	8.0414	3.6220	1.6144	8.0631	3.9010	1.7215	7.3922	3.6500	1.8977	0.8566
0.2554	7.8711	3.4677	1.5661	8.0915	3.9117	1.6972	6.9003	3.1556	1.6047	0.7415
0.3193	7.9260	3.4573	1.5415	7.9471	3.8230	1.6314	6.2012	2.5197	1.2201	0.5775
0.4257	7.8905	3.4783	1.4978	7.4512	3.5699	1.5036	5.4363	2.0254	0.9286	0.4267
0.5108	7.5331	3.3857	1.4312	6.9245	3.3010	1.3766	4.8553	1.7174	0.7593	0.3483
0.6386	6.7207	3.1282	1.2900	5.9299	2.7753	1.1435	3.9840	1.3194	0.5498	0.2529
0.8514	5.3573	2.5627	0.9961	4.0316	1.7641	0.7061	2.4191	0.7048	0.2588	0.1198
1.0217	4.4616	2.1622	0.8350	3.1272	1.2524	0.4648	1.4429	0.3614	0.1314	0.0662
1.2772	3.6308	1.7051	0.6308	2.2112	0.7920	0.2429	0.5905	0.1109	0.0419	0.0231
1.7029	2.6365	1.1341	0.3890	1.3237	0.4622	0.1246	0.2726	0.0347	0.0089	0.0050
2.0435	1.8047	0.7022	0.2250	0.7143	0.2255	0.0482	0.0952	0.0106	0.0028	0.0025
2.5545	0.9690	0.3363	0.1032	0.2944	0.0653	0.0059	0.0018	0.0008	0.0008	0.0008
3.4059	0.3418	0.1032	0.0318	0.1073	0.0191	0.0008	—	—	—	—
4.0871	0.1287	0.0350	0.0095	0.0431	0.0097	0.0004	—	—	—	—
5.1089	0.0038	0.0011	0.0003	0.0015	0.0003	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-4}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-8}$)
0.0340	6.2001	3.8129	1.5202	3.4875	1.0617	3.7626	1.5102	6.1310	2.8841	9.2033
0.0425	6.4248	3.8784	1.5302	3.4918	1.0695	3.8718	1.5671	6.2943	2.8562	8.3862
0.0567	6.5176	3.8003	1.4758	3.3972	1.0610	3.9241	1.6086	6.4384	2.7745	7.0333
0.0729	6.3209	3.5095	1.3327	3.1271	1.0065	3.8354	1.6223	6.6807	2.8081	6.4104
0.1021	5.5470	2.7280	0.9762	2.4471	0.8492	3.4399	1.5905	7.2579	3.1031	7.0239
0.1277	4.7471	2.0209	0.6642	1.8207	0.6845	2.9289	1.4895	7.4837	3.2457	7.3002
0.1703	3.7811	1.3421	0.3880	1.1096	0.4390	2.0379	1.2614	7.4432	3.2904	7.3164
0.2128	3.1810	1.0372	0.2799	0.6961	0.2695	1.4195	1.0688	7.0530	3.1415	6.8926
0.2554	2.7146	0.8060	0.1960	0.4243	0.1876	1.2328	0.9718	6.4405	2.8429	6.1356
0.3193	2.0457	0.5203	0.1155	0.2498	0.1319	0.9787	0.7781	5.1029	2.2058	4.5963
0.4257	1.4405	0.3402	0.0840	0.1924	0.0854	0.5215	0.3974	2.5221	1.0163	1.8517
0.5108	1.2277	0.3151	0.0821	0.1713	0.0590	0.2620	0.1851	1.1318	0.4157	0.5944
0.6386	0.9694	0.2901	0.0810	0.1734	0.0557	0.2092	0.1446	0.9127	0.3451	0.4944
0.8514	0.5640	0.2295	0.0725	0.1817	0.0605	0.2182	0.1488	0.9325	0.3431	0.4413
1.0217	0.3589	0.1698	0.0600	0.1817	0.0622	0.2123	0.1420	0.8841	0.3182	0.3737
1.2772	0.1378	0.0772	0.0374	0.1685	0.0600	0.1861	0.1203	0.7470	0.2662	0.2986
1.7029	0.0358	0.0288	0.0221	0.1314	0.0448	0.1035	0.0584	0.3625	0.1292	0.1450
2.0435	0.0228	0.0200	0.0153	0.0896	0.0279	0.0359	0.0111	0.0685	0.0244	0.0274
2.5545	0.0076	0.0067	0.0052	0.0303	0.0092	0.0080	—	—	—	—
3.4059	—	—	—	0.0001	—	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 11. The data of Table 9 with volatile constituents removed

Energy (MeV)	31.2 min ($\times 10^{-1}$)	1.12 hr ($\times 10^{-1}$)	2.40 hr ($\times 10^{-1}$)	5.16 hr ($\times 10^{-2}$)	11.1 hr ($\times 10^{-2}$)	23.8 hr ($\times 10^{-2}$)	2.13 days ($\times 10^{-3}$)	4.57 days ($\times 10^{-3}$)	9.82 days ($\times 10^{-3}$)	21.1 days ($\times 10^{-3}$)
0.0340	4.9612	2.4273	0.8560	2.8571	1.4344	0.8284	5.2642	2.9682	1.3742	0.6915
0.0425	5.4059	2.6334	0.9304	3.1352	1.5639	0.8910	5.6140	3.1557	1.4660	0.7370
0.0567	5.9766	2.8890	1.0242	3.5101	1.7318	0.9636	5.9769	3.3408	1.5623	0.7846
0.0729	6.4393	3.0834	1.0978	3.8423	1.8718	1.0116	6.1509	3.4134	1.6110	0.8093
0.1021	6.9175	3.2504	1.1681	4.2653	2.0313	1.0365	6.0413	3.3017	1.5925	0.8053
0.1277	7.0710	3.2615	1.1839	4.5094	2.1076	1.0184	5.6791	3.0538	1.5113	0.7761
0.1703	6.9779	3.1091	1.1532	4.7621	2.1706	0.9570	4.9033	2.5545	1.3396	0.7282
0.2128	6.6695	2.8665	1.0937	4.9375	2.2296	0.9192	4.3755	2.2076	1.2035	0.6781
0.2554	6.3766	2.6622	1.0460	5.1041	2.3172	0.9293	4.2619	2.0962	1.1250	0.6165
0.3193	6.3984	2.6620	1.0582	5.2953	2.3951	0.9302	4.0435	1.8948	0.9820	0.5137
0.4257	6.4501	2.7597	1.0882	5.3201	2.3839	0.8919	3.5844	1.5631	0.7824	0.3929
0.5108	6.1901	2.7198	1.0618	5.0884	2.2834	0.8466	3.2572	1.3555	0.6673	0.3329
0.6386	5.5586	2.5330	0.9641	4.4480	2.0209	0.7572	2.7606	1.0521	0.4971	0.2483
0.8514	4.6142	2.1422	0.7642	3.1798	1.4361	0.5451	1.7707	0.5344	0.2266	0.1171
1.0217	4.0287	1.8688	0.6620	2.5677	1.0801	0.3814	1.0769	0.2568	0.1120	0.0644
1.2772	3.4181	1.5394	0.5308	1.9317	0.7443	0.2271	0.5039	0.0715	0.0305	0.0221
1.7029	2.5011	1.0359	0.3327	1.1649	0.4403	0.1209	0.2513	0.0251	0.0061	0.0047
2.0435	1.6965	0.6279	0.1849	0.6097	0.2141	0.0475	0.0925	0.0101	0.0027	0.0025
2.5545	0.9077	0.3015	0.0876	0.2624	0.0639	0.0059	0.0018	0.0008	0.0008	0.0008
3.4059	0.3243	0.0960	0.0297	0.1048	0.0194	0.0008	—	—	—	—
4.0871	0.1242	0.0337	0.0094	0.0437	0.0099	0.0004	—	—	—	—
5.1089	0.0037	0.0011	0.0003	0.0016	0.0003	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-4}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-8}$)
0.0340	4.7722	3.2878	1.4393	3.4239	1.0393	3.6970	1.4824	6.0310	2.8728	9.2019
0.0425	4.9747	3.3536	1.4550	3.4721	1.0634	3.8232	1.5368	6.1842	2.8438	8.3846
0.0567	5.0880	3.2890	1.4024	3.3770	1.0541	3.8693	1.5475	6.3143	2.7605	7.0315
0.0729	4.9875	3.0406	1.2654	3.1072	0.9989	3.7750	1.5846	6.5439	2.7927	6.4085
0.1021	4.4740	2.3662	0.9240	2.4288	0.8408	3.3720	1.5482	7.1042	3.0858	7.0217
0.1277	3.9245	1.7586	0.6261	1.8042	0.6755	2.8565	1.4444	7.3198	3.2272	7.2978
0.1703	3.3829	1.2433	0.3731	1.0965	0.4293	1.9604	1.2131	7.2677	3.2707	7.3139
0.2128	3.0316	1.0279	0.2777	0.6850	0.2596	1.3403	1.0195	6.8737	3.1213	6.8901
0.2554	2.5990	0.7987	0.1942	0.4135	0.1779	1.1554	0.9235	6.2652	2.8232	6.1331
0.3193	1.9687	0.5139	0.1138	0.2401	0.1233	0.9093	0.7348	4.9457	2.1881	4.5941
0.4257	1.3992	0.3360	0.0829	0.1860	0.0796	0.4753	0.3686	2.4168	1.0045	1.8502
0.5108	1.2117	0.3129	0.0815	0.1679	0.0560	0.2377	0.1700	1.0767	0.4095	0.5936
0.6386	0.9687	0.2899	0.0809	0.1732	0.0555	0.2079	0.1438	0.9096	0.3448	0.4943
0.8514	0.5640	0.2295	0.0725	0.1817	0.0605	0.2182	0.1488	0.9325	0.3431	0.4413
1.0217	0.3589	0.1698	0.0600	0.1817	0.0622	0.2123	0.1420	0.8841	0.3182	0.3737
1.2772	0.1378	0.0772	0.0374	0.1685	0.0600	0.1861	0.1203	0.7470	0.2662	0.2986
1.7029	0.0358	0.0288	0.0221	0.1314	0.0448	0.1035	0.0584	0.3625	0.1292	0.1450
2.0435	0.0228	0.0200	0.0153	0.0896	0.0279	0.0359	0.0111	0.0685	0.0244	0.0274
2.5545	0.0076	0.0067	0.0052	0.0303	0.0092	0.0080	—	—	—	—
3.4059	—	—	—	0.0001	—	—	—	—	—	—
4.0871	—	—	—	—	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

Table 12. *The data of Table* 10 *with volatile constituents removed*

Energy (MeV)	31.2 min ($\times 10^{-1}$)	1.12 hr ($\times 10^{-1}$)	2.40 hr ($\times 10^{-1}$)	5.16 hr ($\times 10^{-2}$)	11.1 hr ($\times 10^{-2}$)	23.8 hr ($\times 10^{-2}$)	2.13 days ($\times 10^{-3}$)	4.57 days ($\times 10^{-3}$)	9.82 days ($\times 10^{-3}$)	21.1 days ($\times 10^{-3}$)
0.0340	4.6291	2.1916	0.7890	2.7965	1.4393	0.8280	5.2336	2.9380	1.3627	0.6903
0.0425	5.0532	2.3810	0.8586	3.0708	1.5700	0.8912	5.5850	3.1250	1.4542	0.7357
0.0567	5.6045	2.6188	0.9472	3.4422	1.7400	0.9648	5.9530	3.3111	1.5505	0.7834
0.0729	6.0622	2.8039	1.0180	3.7734	1.8826	1.0143	6.1355	3.3868	1.5998	0.8081
0.1021	6.5632	2.9752	1.0892	4.2001	2.0466	1.0422	6.0453	3.2839	1.5837	0.8044
0.1277	6.7600	3.0055	1.1100	4.4514	2.1268	1.0266	5.7015	3.0452	1.5052	0.7754
0.1703	6.7660	2.9038	1.0928	4.7195	2.1948	0.9689	4.9524	2.5600	1.3378	0.7279
0.2128	6.5647	2.7198	1.0486	4.9097	2.2563	0.9325	4.4365	2.2200	1.2040	0.6780
0.2554	6.3577	2.5644	1.0130	5.0859	2.3436	0.9420	4.3186	2.1072	1.1255	0.6165
0.3193	6.4120	2.5860	1.0287	5.2774	2.4199	0.9414	4.0915	1.9030	0.9824	0.5137
0.4257	6.4437	2.6831	1.0556	5.2958	2.4062	0.9014	3.6221	1.5683	0.7826	0.3929
0.5108	6.1621	2.6417	1.0287	5.0626	2.3045	0.8559	3.2940	1.3602	0.6675	0.3329
0.6386	5.5039	2.4571	0.9326	4.4227	2.0395	0.7661	2.7962	1.0567	0.4973	0.2483
0.8514	4.5572	2.0828	0.7406	3.1608	1.4489	0.5524	1.8015	0.5387	0.2267	0.1171
1.0217	3.9950	1.8297	0.6471	2.5575	1.0891	0.3871	1.1015	0.2605	0.1103	0.0644
1.2772	3.4136	1.5235	0.5250	1.9305	0.7496	0.2307	0.5191	0.0741	0.0306	0.0221
1.7029	2.5045	1.0291	0.3297	1.1622	0.4426	0.1227	0.2583	0.0262	0.0061	0.0047
2.0435	1.7066	0.6269	0.1840	0.6073	0.2146	0.0481	0.0946	0.0103	0.0027	0.0025
2.5545	0.9218	0.3038	0.0878	0.2607	0.0634	0.0059	0.0018	0.0008	0.0008	0.0008
3.4059	0.3313	0.0972	0.0297	0.1036	0.0190	0.0008	—	—	—	—
4.0871	0.1268	0.0341	0.0094	0.0431	0.0097	0.0004	—	—	—	—
5.1089	0.0038	0.0011	0.0003	0.0015	0.0003	—	—	—	—	—

Energy (MeV)	45.3 days ($\times 10^{-4}$)	97.3 days ($\times 10^{-4}$)	208 days ($\times 10^{-4}$)	1.20 years ($\times 10^{-5}$)	2.60 years ($\times 10^{-5}$)	5.58 years ($\times 10^{-6}$)	12.0 years ($\times 10^{-6}$)	25.7 years ($\times 10^{-7}$)	55.3 years ($\times 10^{-7}$)	119 years ($\times 10^{-8}$)
0.0340	4.7721	3.2878	1.4393	3.4239	1.0393	3.6970	1.4824	6.0310	2.8728	9.2019
0.0425	4.9746	3.3536	1.4550	3.4721	1.0634	3.8232	1.5368	6.1842	2.8438	8.3846
0.0567	5.0879	3.2890	1.4024	3.3770	1.0541	3.8693	1.5745	6.3143	2.7605	7.0315
0.0729	4.9874	3.0406	1.2654	3.1072	0.9989	3.7750	1.5846	6.5439	2.7927	6.4085
0.1021	4.4739	2.3662	0.9240	2.4288	0.8408	3.3720	1.5482	7.1042	3.0858	7.0217
0.1277	3.9245	1.7586	0.6261	1.8042	0.6755	2.8565	1.4444	7.3198	3.2272	7.2978
0.1703	3.3829	1.2433	0.3731	1.0965	0.4293	1.9604	1.2131	7.2677	3.2707	7.3139
0.2128	3.0316	1.0279	0.2777	0.6850	0.2596	1.3403	1.0195	6.8737	3.1213	6 8901
0.2554	2.5990	0.7987	0.1942	0.4135	0.1779	1.1554	0.9235	6.2652	2.8232	6.1331
0.3193	1.9687	0.5139	0.1138	0.2401	0.1233	0.9093	0.7348	4.9457	2.1881	4.5941
0.4257	1.3992	0.3360	0.0829	0.1860	0.0796	0.4753	0.3686	2.4168	1.0045	1.8502
0.5108	1.2117	0.3129	0.0815	0.1679	0.0560	0.2377	0.1700	1.0767	0.4095	0.5936
0.6386	0.9687	0.2899	0.0809	0.1732	0.0555	0.2079	0.1438	0.9096	0.3448	0.4943
0.8514	0.5640	0.2295	0.0725	0.1817	0.0605	0.2182	0.1488	0.9325	0.3431	0.4413
1.0217	0.3589	0.1698	0.0600	0.1817	0.0622	0.2123	0.1420	0.8841	0.3182	0.3737
1.2772	0.1378	0.0772	0.0374	0.1685	0.0600	0.1861	0.1203	0.7470	0.2662	0.2986
1.7029	0.0358	0.0288	0.0221	0.1314	0.0448	0.1035	0.0584	0.3625	0.1292	0.1450
2.0435	0.0228	0.0200	0.0153	0.0896	0.0279	0.0359	0.0111	0.0685	0.0244	0.0274
2.5545	0.0076	0.0067	0.0052	0.0303	0.0009	0.0008	—	—	—	—
3.4059	—	—	—	—	0.0092	0.0080	—	—	—	—
4.0871	—	—	—	0.0001	—	—	—	—	—	—
5.1089	—	—	—	—	—	—	—	—	—	—

BETHE–BACHER[10] approximation of FERMI's[11] exact theoretical solution.

The general procedure for calculating the distributions of beta rays is similar to the method described for γ-rays. Here, however, the spectra are continuous and the calculation is made at

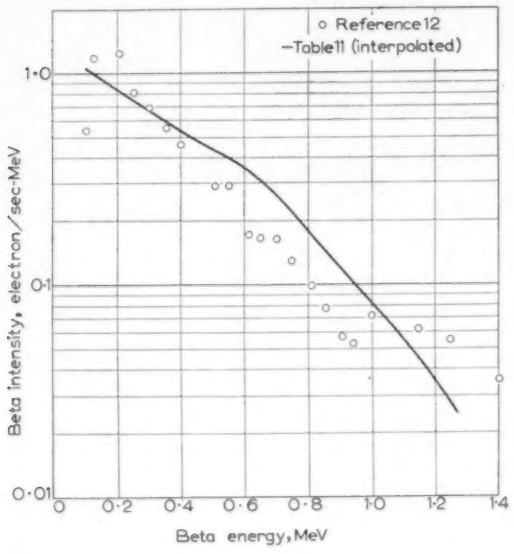


FIG. 3. Comparison of the calculated β-spectra from the yield theory of GLENDENIN and experimental β-spectra 18 days after fission.

exact energies instead of averaging over finite energy boxes. The number of electrons with energy between τ and $\tau + d\tau$ emitted by nuclide j may be expressed by $\sum_n S_{n,j}(\tau)\, d\tau$, where n indicates a specific mode of disintegration. The β-abundance (number of β-particles/disintegration) of nuclide j at energy τ is given by:

$$N_j(\tau) = \sum_n \frac{S_{n,j}(\tau)}{\int_0^{\tau_n} S_{n,j}(\tau')\, d\tau'} \times R_{j,n}$$

$$0 < \tau \leq \tau_n \quad (2)$$

where the first factor represents the relative number of β-particles in the spectrum with energy τ and $R_{j,n}$ is the branching ratio of the nth transition, $\sum_n R_{j,n} = 1$.

The total number of β-particles with energy τ at time t_i is written:

$$K(t_i, \tau) = \sum_j N_j(\tau) L_j(t_i) \quad (3)$$

where $L_j(t_i)$ is defined in Section (2a).

(b) *Results*

The main tabulation from equation (3), is given for alternative sets of input data namely, GLENDENIN[1b] and PRESENT.[3] Tables 9 and 10 include all β-emitting nuclides which are listed in Table 8. The results calculated by excluding β-rays from volatile materials (Br, Kr, Ru, I and Xe) appear in Tables 11 and 12. For uniformity the energies and times selected correspond to those used in the γ-ray tables.

(c) *Comparison with experiment*

Interpolated results of Table 9 have been compared with the experimental β-spectrum of "Bikini ash"[12] in Fig. 3. This figure shows only the similarity in shape of the curves because the intensity of the Bikini data cannot be compared with the calculated data on an absolute scale.

REFERENCES

1. C. D. CORYELL and N. SUGARMAN (Editors), *Radiochemical Studies: The Fission Products*. National Nuclear Energy Series, Book I, Division IV, Part IV, Papers 34 and 52. McGraw-Hill, New York (1951).
   (a) L. B. BORST, *Estimates of Amount of Radiation and of Accompanying Energy Liberation from Fission Products* Paper 34, p. 344.
   (b) L. E. GLENDENIN, C. D. CORYELL and R. R. EDWARDS, *Distribution of Nuclear Charge in Fission* Paper 52, p. 489.
2. R. C. BOLLES and N. E. BALLOU, *Calculated Activities and Abundances of* U^{235} *Fission Products*. U.S. Naval Radiological Defense Laboratory Report USNRDL-456 (1956).
3. R. D. PRESENT, *Phys. Rev.* **72,** 7 (1947).
4. C. F. MILLER, *Gamma Decay of Fission Products from the Slow-neutron Fission of* U^{235}. U.S. Naval Radiological Defense Laboratory Report USNRDL-TR-187 (1957).

5(a). J. MOTEFF, *Fission Product Decay Gamma Spectrum*. G.E. Aircraft Nuclear Propulsion Project APEX 134 (1953).

5(b). F. H. CLARK, *Decay of Fission Product Gammas*. Report No. NDA-27-39 (1954).

6. J. F. PERKINS and R. W. KING, *Nucl. Sci. Engng.* **3,** 726 (1958).
7. W. ZOBEL and T. A. LOVE, *Short Lived Fission Product Gamma Radiation.* Paper presented at the Symposium of The Shorter-Term Biological Hazards of a Fallout Field, Washington, D.C. (1956).
8. J. M. WYCKOFF and H. W. KOCH, Private communication.
9. *Tables for the analysis of beta spectra*, National Bureau of Standards Applied Math Series-13 (1952); M. E. ROSE, Theory of allowed beta decay. *Beta and Gamma Ray Spectroscopy* (Edited by K. SIEGBAHN). Interscience Publishers, New York (1955).
10. H. A. BETHE and R. F. BACHER, *Rev. Mod. Phys.* **8,** 194 (1936).
11. E. FERMI, *Z. Physik* **88,** 161 (1934).
12. The Committee for Compilation of Reports on Research in the Effects of Radioactivity, *Research in the Effects and Influences of the Nuclear Bomb Test Explosions.* Japan Society for the Promotion of Science, Tokyo (1956); Y. NISHIWAKI *et al.*, *Studies on the Radioactivity of Bikini Ash*, pp. 457–469.

Health Physics Pergamon Press 1959. Vol. 1, pp. 442–446. Printed in Northern Ireland

COMMENTS ON THE LABOR RELATIONS ASPECT OF NUCLEAR ENERGY*

C. F. MacGOWAN
International Representative, International Brotherhood of Boilermakers, Iron Ship Builders, Blacksmiths, Forgers and Helpers, 1837 Crestwood Avenue, Munster, Indiana

(*Received* 7 *July* 1958)

Abstract—The objective of this paper is to establish that the preferred solution to questions of labor relations in the field of nuclear energy is to anticipate and render those questions unnecessary by means of a good safety (and health physics) program.

Discussed herein are contract provisions covering safety programs, methods of their application and some of the variables which may be introduced into this relatively new area, as well as the necessity for labor to adopt certain viewpoints with respect to certain existing problems and recommendations which have been made by various advisory groups.

ONE of the least discussed and hence possibly one of the least understood aspects of safety in nuclear employment is that of labor relations and of how the subject of labor relations may have a bearing on that of safety. We have also found the converse to be true, for safety may have an important bearing on the current or future course of such relations.

To the writer's knowledge there has never been any real disagreement between management and labor as to the desirability of a successful safety program, regardless of whether the program originates with management or with labor. It is axiomatic that, regardless of the type of industry, nuclear or not, the inauguration of or maintenance of a system of sound safety practices is not universally or unilaterally the responsibility of management alone. It is usually the result of co-operative effort by both management and its employees. While it is not possible to assert that *all* joint programs are successful, nevertheless *all successful* programs *are* jointly sponsored and administered.

Since collective bargaining is the yardstick by which most labor relations are measured, it seems logical to start with a discussion of nuclear safety aspects which are concerned with that process, and consider later some of the problems which are incident to or resulting from such negotiations.

For the purpose of this discussion then, this paper will be empirically divided into (1) direct labor relations aspects or collective bargaining; and (2) indirect relations which affect that process.

Direct relations

There are several ways in which safety becomes a part of a collective bargaining agreement, but basically they can be grouped into two subdivisions: (a) bargaining agreements with formal safety committees and specific programs; and (b) agreements with more generalized provisions.

Taking these two types of safety agreements in order, and as a measure of the relative uses thereof, a recent (1954–1955) survey[1] by the U.S. Department of Labor of nearly 1600 major collective bargaining agreements (each of which covered 1000 or more workers) found that nearly 22 per cent of these agreements, covering over two million people, contained provision for formal joint safety committees

* President's Conference on Occupational Safety. Workshop session: Radiation, a controllable hazard.

and programs. A far greater number, in fact most others, cover the subject less formally, with arrangements which are a definite part of and negotiated by means of collective bargaining, but having no specific provisions or committees which must channel safety measures in a definitely prescribed way.

Formal committees, with specific programs, are found most often in mining or in other industries where, with certain recognized hazards, it is possible to draw up in the bargaining agreements specific provisions relating to those hazards, with mutual recognition of codes applicable. It is in the area of this type of agreement that most nuclear safety bargaining clauses will be found.

Examples of the more generalized provisions are: (a) stated pledges by management to observe applicable standards recommended or required by government or industry advisory groups (this will be the second most popular type of nuclear safety provision); (b) agreement that necessary protective devices will be supplied; and (c) that satisfactory ventilation, light, heat, etc., will be supplied for healthful working conditions. Note that in the latter types no allusions are made to anything specific, but the fields covered are general enough that by mutual understanding they can be extended to cover any aspect of nuclear technology.

The advent of industrial uses of nuclear energy then, poses no *new* problems as to the methods of negotiating for and putting into effect, an effective safety program since we already have these well established methods. We are able to utilize specific recommendations and references as other industries have done because of the activities of the National Committee on Radiation Protection and Measurement. Indeed, at this time it seems most probable that the recommendations of that respected Committee will ultimately be used as a basis for, if not physically incorporated into, most collective bargaining agreements in the field of nuclear energy.

Worthy of mention is the fact that the activities of this Committee, together with its international counterpart the International Commission on Radiological Protection, provide a basis for international standards, with the probability that these standards will in turn be recognized as a basis for labor relations worldwide, one of the very few to be so accepted. While many of the aspects of nuclear energy employment are inherently dangerous to health and well-being, the fact is that these can be controlled within rather narrow limits. SMITH,[4] AEC's Director of Organization and Personnel, whose supervisory responsibility includes that of the Safety and Fire Protection Branch, stated recently that ". . . the injury frequency rate . . . for contractor operations reached an all time low in the program of 1.62 about a year ago. It has now risen to 1.89. This, however, represents an improvement of 0.72 since 1953, and compares favorably with an injury frequency rate for manufacturing generally of 4.42; or an injury frequency rate for the petroleum industry of 7.26 or the chemical industry of 3.38 . . . if the injury frequency rate is a valid indicator, it appears that employment in the government atomic energy work is among the safest in all industry." AEC's severity rate was 229 as compared with Petroleums 861 and Chemical's 462 for the year previous.

Since work in the government atomic energy programs most nearly resembles or parallels the type of work done in chemical, refining and metals industries, it may seem paradoxical that an industry which is only little more than 10 years old is able to evidence such comparatively low rates, particularly since so much of the work is largely experimental.

Part of the answer lies, possibly, with the fact that safety in nuclear work, because of its uncertainties, is a dynamic thing and because of these very uncertainties together with the permanence of the results of radiation injuries, much more emphasis (and money) is placed upon it than in other industry. Probably, while the government remains in its supervisory position this will continue to be true.

Labor, looking as it must into the future, foresees the possibility that this situation might not remain true when industry alone holds the rein. Putting it another way, we feel that a problem will be posed by those companies who are engaged in the production or utilization of radiation or its sources on a marginal basis.

It is entirely possible that those companies which are not in the business in a "big way" will, by reason of simple economic necessity, be tempted to dispense with part or all of the costly and non-productive overhead of health physics programs and related equipment. Certainly, if business opportunities expand as has been forecast, not all companies in it can be big, nor can the operations of any of them, large or small, become automatically and immediately profitable enough to disregard costly investments in services and equipment. It is at this point that collective bargaining agreements may prove their worth with respect to safety.

The following should be clear. That segment of labor with which the writer is associated has no intention or making the utilization or production of nuclear components economically prohibitive by unnecessarily restrictive practices and bargaining provisions. We do, however, intend to do our utmost to assure our members of working conditions which are as safe as modern knowledge and experience permit. There is no monetary value for human life, and no fair premium that may be offered for working under hazardous conditions—the need is, rather, to establish the standards of protection and to enforce the use of protective devices by agreement, if necessary, that will minimize the risks of injury. Since this is almost precisely the viewpoint of any safety program, it is probably true that difficulties will arise, not from this principle or its application, but rather from the mechanics of operating a program.

Much emphasis is placed upon the problems of nuclear energy by most of the National Unions who are affected by it. Five such Internationals are members of the Atomic Industrial Forum. Five others have Access Permits with staff members assigned thereto. Three others are considering the advisability of such permits in their individual cases, and have already assigned members of their staffs to study the situation. It is the desire of these National Unions to study the problems of the field, in order to be in a position to counsel and advise upon matters affecting health, safety, technology, economic and labor relations.

Indirect relations

In any consideration of the indirect effects of nuclear safety on labor relations, we must include those facets of labor relations which, though perhaps not a direct result of collective bargaining are nevertheless tangent to it; meaning, of course, that at some point in their development they are affected one way or another by negotiations.

One example of these tangential aspects is that of Apprenticeship Training. We (the Boilermakers International) have negotiated with employers covering a majority of our members for a new Apprenticeship Plan made necessary by advancing technology. The new plan will include a sound introduction to the subject of radiation; its nature; types and uses of detection equipment; protective devices and work habits made necessary by it.

While we do not expect to make health physicists of each apprentice, we *do* hope to achieve a nucleus of people who have some authentic knowledge of the subject which we hope will be imparted to others. In addition, we hope to attract the interest of those mechanics who are not a part of the program so they will join for the radiation phases of the course. The writer's international and others like it are sincerely endeavoring to make a contribution to the art. This may be one means by which this is to be accomplished.

Another indirect phase which may be encountered in the near future concerns the growing sentiment for providing a record of personal X-rays in conjunction with in-plant exposures for those employees whose work brings them in contact with radiation. It is our contention that only a thorough knowledge of the whole picture of an employees' radiation history, including that received medically, is a valid indicator of his relative safety from overexposure under certain conditions. ASA Nuclear Standards Subcommittee N2.2 is undertaking the preparations of record standards on this basis.

Labor's view, which is necessarily more confined than that of other groups, is concerned principally with the individual. Labor feels, consequently, that such records must concern that individual's personal radiation status as

well as that of his group, and that such records afford the only means of making this determination. While it is gratifying to note that more and more employers have begun such a record system, nevertheless, it would come as no surprise to find this subject appearing as an item for collective bargaining.

While this next subject may at first appear to be extraneous it can be tied in with the subject of labor relations quite easily. That is the subject of Workmen's Compensation, to which the AFL-CIO and its affiliates have alluded time and again, testifying as to the importance of revisions in the several laws to the Joint Committee on Atomic Energy and other law-making bodies. Interest is awakening for, in the past year, several advisory commissions and boards have reported on the subject.

From a comparison of the findings of these committees there has appeared, among others, one important fact: The apparent trend toward deferring to the Federal Government or Federal standards in the matter of State vs. Federal control. For instance, the National Planning Association,[10] recommended that: "Federal Standards be considered in compensation cases involving radiation injuries or sickness." In December 1957, the Joint Federal–State Action Committee[9] and the Conference of State Governors recommended amendments to the Atomic Energy Act of 1954 giving authority to states in certain areas not in conflict with federal standards.

The Atomic Energy Committee of the International Association of Industrial Accident Boards commented[10] that: "... the public will demand Workmen's Compensation coverage of radiation injuries on a Federal level in the event the States are laggardly in revising their present Compensation provisions." Note: labor has for several years found that this "laggardly" attitude exists. CHEITE[14] reported that only about nineteen States had the bare necessities of basic radiation coverage in their laws.

In setting forth its suggestions for rectifying the laws, the Association of Industrial Accident Boards' Committee made five recommendations: (a) coverage of *all* radiation cases; (b) revised statutes of limitations; (c) unlimited medical benefits; and (d) making radiation injuries eligible for second injury benefits.

The Committee's fifth point, however, is very unusual and unexpected. It recommends that the states have the authority to designate physicians and hospitals for radiation cases. It seems an ironic twist in these days of the great debate over "socialized medicine" for this committee to go beyond anything yet advocated by its proponents; and for labor to be among the first to point out that the victim of radiation injuries would thus become a trophy in what promises to be a monumental government tug-of-war.

This discussion of Workmen's Compensation deficiencies is intended to point up one significant fact of labor relations. Serious problems, many of which are mutual with both management and labor, have a habit of appearing at the bargaining table. As ample evidence of this it was the inadequate means of meeting the economic needs of unemployed persons which created the climate required to provide impetus to Supplementary Unemployment Benefit bargaining which has proved itself and is now, though still inadequate, a mainstay for many recession-struck localities.

General

Labor relations, of course, provide no panacea for all the situations which will arise in safety programs. Neither Labor nor anyone else knows all the answers. We find ourselves speculating about the recent action by the AEC[5] which approved for use in its facilities and those of its contractors the newer NCRPM recommendations, but excluded its licensees, thus destroying the alignments of parallel exposure limits which have existed for years in both types of employment. We wonder about genetic effects, about safety programs, about "let-downs" in safety standards.

We realize that, as AFL-CIO[12] puts it; "Even with the best intentions, with its importance acknowledged by both management and labor, safety procedure lapses into the routine, with its sensitivity blunted ... and that every union can benefit from a fresh look at its safety activities."

We have learned that by gaining constant

management concern with accident prevention, worker awareness of the need for safe practices and regular review and inspection of hazardous situations, unions can make a contribution to safety, especially in nuclear industry.

Three years ago there was almost no allusion to nuclear hazards in the bargaining agreements extant in the industry. Since then a survey by the AFL-CIO has shown that over 60 per cent of agreements in similar but non-nuclear industries such as metals, chemicals and construction, have some specific provision for safety. Thus, with a growing nuclear industry the fact must be recognized that collective bargaining agreements in this field will assume a similar and major position of importance.

REFERENCES

1. National Safety Council Facts, (1956).
2. Accident at Windscale No. 1 Pile. Statement by Prime Minister on 10th October, 1957.
3. *Radiation Safety and Major Activities in Atomic Energy Programs.* USAEC 1/57.
4. S. Smith, Address given on May 7, 1957.
5. AEC Press Release No. 1231. (1957).
6. NBS Addendum to Handbook 60.
7. AFL-CIO Monthly Report. (1957).
8. *Report by Atomic Energy Committee on International Association of Industrial Accident Boards and Commissions.* (1957).
9. *Report of Joint Federal-State Action Committee and Conference of State Governors.* (1957).
10. *Report of National Planning Association.* (1957).
11. *USAEC Report of Disabling Injury Experience of Contractors and AEC Offices.* (1957).
12. *Report of the AFL-CIO Executive Council.* (*Second Convention*), (1957).
13. D. F. Haynes, *Summary of Incidents Involving Radioactive Material in Atomic Energy Activities, January–December,* 1956.
14. F. Cheite, *AFL-CIO Atomic Energy Conference Papers.* (1957).

Health Physics Pergamon Press 1959. Vol. 1. pp. 447–448. Printed in Northern Ireland

NOTE

A Portable Calibrator for β–γ Survey Instruments using Sr^{90}

(*Received* 31 *October* 1958)

INTRODUCTION

A NEW method of calibrating portable radiation detection instruments was needed at Atomics International by which an instrument could be calibrated on a bench near an instrument repair function without the constant use of a calibration facility. Such a device or method would mean easier instrument maintenance and standardization, a much smaller and less expensive γ-calibration facility, and little or no radiation to the calibration personnel; furthermore, the accessibility would increase the feasibility of daily or weekly instrument calibration checks. Such a calibrator cannot be compared to an internal β-source which checks one point on the scale, since a complete range of intensities are available.

The problem was to construct a device (with an appropriate source) which would meet the majority of the objectives.

APPARATUS AND MATERIALS

Fig. 1 shows the construction diagram of the first model. A 6 in. diameter steel tube $\frac{1}{4}$ in. thick was used as the outer housing for the Sr^{90} source, floating table, threaded products, drives and limit switches. The housing for the instrument being calibrated was made from $\frac{1}{2}$ in. plexiglass. The indicator, located on top of the plexiglass housing, gives the revolutions of the threaded product or the distance between the source and the instrument. The source consists of 3 mc of Sr^{90}–Y^{90} with a 2 mil stainless steel window. The calibrator was constructed with a particular height and shape such that it could be made an integral part of the working bench or used as a special calibration table. The operational portion of the calibrator is exactly bench height for maximum accessibility.

A reversible 180 rev/min motor turns the right angle drives which in turn rotate the two threaded rods. The floating table is threaded so that as the rods turn the table rises or lowers, depending upon the direction of rotation. The source is attached to the floating table and is located in the center of the opening in the instrument housing.

The threaded rods are $\frac{1}{4}$ in. in diameter and have 20 threads/in. Since the revolution counter is calibrated in tenths of a revolution, the source distance can be reproduced within 5/1000 of an inch. The rods turn 180 rev/min and the time involved for traveling the 30 in. distance is 3.3 min.

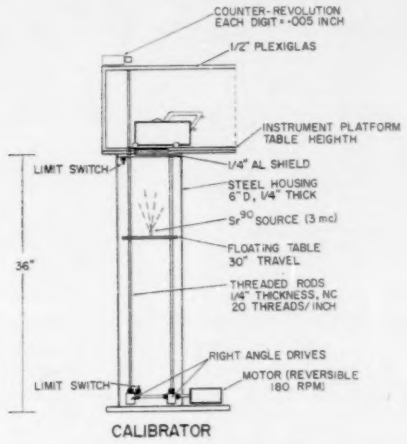


FIG. 1. Calibrator.

The limit switches located near the ends of travel remove the possibility of jamming the mechanism in the cylinder.

When the revolution counting dial is adjusted to a position corresponding to a known response for a particular type of instruments, the sliding shield is pulled to its open position and the radiation detection instrument obtains a specific dosage. If an instrument adjustment is necessary the shield can be placed in the closed position and the adjustment made without removing the instrument and without exposure to personnel. Adaptors were made for various types of instruments such that the geometry could be duplicated easily.

METHOD

To obtain a γ-calibration curve for a particular type of β–γ detection instrument, the instrument was initially calibrated with a standard γ-source and secondly placed in the calibrator. The revolution vs dosage curve is then plotted as illustrated in Fig. 2. The Sr^{90} β-source is now used as a check

source. This method uses the assumption that any β–γ detection instrument which has initially been calibrated with a gamma standard can be maintained in calibration by comparing the readings of the

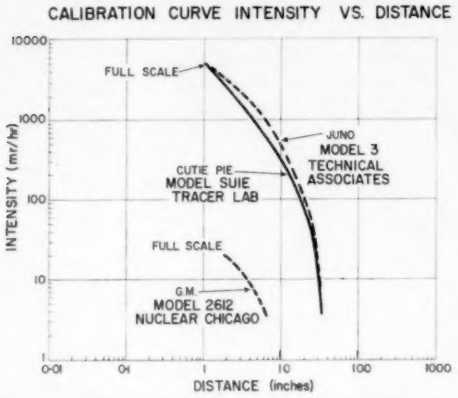


FIG. 2. Calibration curve intensity vs. distance.

instrument obtained at various distances with the Sr^{90} source. As long as these readings can be duplicated the instrument will remain in calibration. Each range on the instrument can be checked for one reading and thus insure complete linearity on all scales. Logarithmic meters are easily calibrated because of the large range of dosage rates available.

If desirable, instruments can be used to show a more precise representation of dose due to β. An extrapolation chamber can be used in conjunction with the calibrator to determine the true yield of ionization from the Sr^{90}–Y^{90} source and thus determine the β-dose. By this method of calibration, an instrument can determine r/hr with the normal scale reading and with the proper correction factor, rad/hr. The curves in Fig. 2 gives a calibration plot for a low range Juno, Cutie Pie, and G.-M. survey meter. The curve deviates from a straight line at close distances because of the nearness of the Sr^{90} source to the large chamber volumes and falls rapidly because of the greater absorption of the beta particles in air at the longer distances.

DISCUSSION OF RESULTS

Instruments of a particular type were calibrated with radium, placed in the calibrator, and the corresponding curves were plotted as discussed above. It was found that for all practical purposes the data from all the instruments for a particular type fell on the same curve and that these curves were reproducible.

The calibrator has been in use for nearly a year and it has been found that all β–γ portable detection instruments or thin walled γ-instruments which have air equivalent walls can be kept in calibration by its use. The calibrator has fulfilled all the objectives and is inexpensive to build.

SUMMARY

An instrument has been described which has almost become indispensible in calibrating portable radiation detection instruments. It has the following features:

(1) An instrument can be maintained in calibration without the routine use of a γ-calibration facility.
(2) Instruments can be calibrated with negligible radiation exposure to personnel. The calibrators can be safely used in practically any area. The maximum external dosage rate is 0.4 mrem per hr which is due to the bremsstrahlung produced in the walls of the shield.
(3) The calibrator saves time and is inexpensive to build. The unit cost was $200 including the source.

C. T. NELSON

Atomics International
Canoga Park
California

Health Physics Pergamon Press 1959. Vol. 1. pp. 449. Printed in Northern Ireland

LETTER TO THE EDITOR

Suggested Insignia for the Health Physics Society

(*Received* 1 *November* 1958)

THE above design is submitted for consideration as the insignia of the Health Physics Society.

The classic Health Physics concepts of time, distance and shielding are included in obvious symbolism, and the internationally recognized radiation symbol is incorporated.

J. V. NEHEMIAS

National Sanitation Foundation
School of Public Health
University of Michigan
Ann Arbor, Michigan

Health Physics Pergamon Press 1958. Vol. 1. pp. 450–453. Printed in Northern Ireland

NEWS

News Editor: SAUL J. HARRIS

Health Physics in the News

THE following items concerning Health Physics have appeared in newspapers, public press or technical journals during the last quarter.

High schools instructed on radiation

Six thousand high schools are being given "radiological defense instrument kits" according to the U.S. Office of Civil and Defense Mobilization "to acquaint students with the general principles underlying radiation phenomena". The kits containing instruments, low-level sources and a manual are furnished free through the state Departments of Education.

Radium dial hazards questioned

Two scientists from the Philadelphia College of Pharmacy and Science suggested that luminous dial wrist watches present radiation hazards to the reproductive organs "several times greater than natural background radiation". The report, based on experiments involving measurements of the radiation from some twenty watches, asked the question as to "whether the small benefit that may be received is worth the hazard".

Criticality during transportation regulated

The AEC has issued a regulation, effective November 3, designed to guard against accidental criticality of fissionable material during shipment. Limits are placed on the amounts of fissionable material that AEC licensees may transport or deliver. The regulations are identified as Title 10, Code of Federal Regulations, Part 71.

Radiation levels found safe

The U.S. Health Service announced that its continuing studies of radiation in air and water showed levels which are considered well within safety concentrations.

Radiation legislation proposed

The Council of State Goverments, in its *Suggested State Legislation–Program for 1959*, outlined four "elements of desirable state radiation protective regulation" including the comment that state regulations "should be in substantial conformity with nationally accepted standards, such as those developed by the NCRP and by the AEC". The proposals also state that "it appears desirable for the state to require that (radiation) sources should be registered with appropriate exemptions". The Council of State Goverments also recommended adoption of legislation to ban shoe-fitting X-ray fluoroscopes, and commented regarding workmen's compensation legislation that satisfactory answers have not as yet been found "to one question of great, perhaps of over-riding, practical importance—the matter of proof of casual relationship between occupational hazard and injury, in the radiation field".

Life-shortening claimed

A workmen's compensation claim was filed in New York State by a 61-year old physicist who claimed that some 36 years of radiation work had resulted in his receiving sufficient radiation dose (1000 r by his own estimates) to shorten his life expectancy by 5 years. The physicist, who had been employed at Roswell Park Memorial Institute, also claimed he had suffered lung damage and finger damage.

Fear therapy suggested

A member of the British Atomic Energy Authority suggested that the "vague public uneasiness" concerning atomic energy risks could be prevented from becoming fear if the risk became understood, the acceptance of it brought material advantage to the individual, and each person could feel that he could prevent his own exposure. The comment was made that "this inability to escape by personal effort" leads to exaggerated fears.

Patient exposure control urged

Gynecology and genetics specialists meeting as a panel of the American College of Surgeons recommended that physicians check carefully into the radiation history of patients and use minimal doses of X-rays in all cases, especially with expectant mothers. X-rays in pregnancy were considered doubly effective, influencing both the expectant mother and the child. One panelist urged that all such examinations should be stopped.

Ex-radium dial painters sought

The AEC is sponsoring three studies designed to locate about 1000 former radium dial painters or persons receiving radium therapy to permit medical studies on the effects of earlier exposures. Argonne National Laboratory is seeking persons who received radium prescriptions, M.I.T. is seeking both radium dial painters and radium patients and the New Jersey Health Department is seeking radium dial painters. Persons having information on individuals who may have been exposed are requested to get in touch with the New York Operations Office of the AEC.

Food radioactivity found small

The Food and Drug Administration announced that after a 2-year test of foods produced both before and after 1945 that there has been no significant increase in their radioactivity content. The highest increase found, which was still considered safe, was in dried tea leaves, with dairy products second, shellfish third and other fish fourth.

Exposures in Yugoslavia

Reports continued to reach the United States that 6 or 7 scientists (varying according to the source of the report) had received over-exposures as a result of an accident at the Yugoslav's Boris Kidric Nuclear Science Institute in Vinca. One was reported to have died after bone marrow grafts were attempted. Bone marrow grafts on four others were reported to have been successful, although the future condition of the grafts is not conclusive. The cause of the over-exposure has not been made public.

Shielding "peep-holes" manufactured

The Corning Glass Works announced that 12 gigantic sight glasses each weighing some 1000 pounds have been ordered by Westinghouse to be installed as part of the radiation shielding in the submarine reactor compartments being built for the Navy. The windows are a combination of Plexiglas and high-density glass and will permit visual inspection of the reactor compartments.

Waste plant begun

The AEC announced that construction had begun on a $6 million pilot plant in Idaho for the calcination of high-level liquid wastes to study methods of reducing these to more compact and storable solids. The new plant is to be completed in January 1960.

Aerial radiological survey started

The AEC and the U.S. Geological Survey announced that aerial surveys around major AEC installations had begun "to obtain information useful to an understanding of the long-term effects of low-level radiation in the enviroment". The survey is also intended to provide data on background levels at the plants in the event of a change in environmental levels.

42nd radium dial painter dies

The Essex County, New Jersey, medical examiner ruled that the death of a woman (age 57) in September was due to chronic radium poisoning as a result of ingestion of radium during dial painting work in the 1920's. The death was reported to be the 42nd attributed to radium dial painting in a New Jersey plant some 30 years ago.

National Committee on Radiation Protection and Measurements

Announcement of New Publications and of Work in Progress

Safe Handling of Bodies Containing Radioactive Isotopes (A Guide for Surgeons, Pathologists, and Funeral Directors), National Bureau of Standards Handbook 65, issued July 10, 1958*. This report is a modified version of Handbook 56, *Safe Handling of Cadavers Containing Radioactive Isotopes*, published in 1953, and includes new information as well as the revised maximum permissible radiation exposure levels developed since then.† Pertinent information is given for the guidance of mortuary and medical personnel involved in the handling and autopsy of bodies containing radioactive materials. Individuals performing emergency surgery or autopsy on bodies which had received large internal therapeutic doses of a radioactive isotope are subject to radiation exposure; the handbook discusses the reduction of this occasional exposure to the permissible level. The report indicates that, at least for the next few years, highly radioactive bodies will be encountered only rarely. The body must be adequately tagged so that radiation danger will be recognized by the surgeon or embalmer. Risk depends largely on the relative locations of the isotope and the operative site, and how recently and how large a dose the patient

* Obtainable from Superintendent of Documents, Government Printing Office, Washington 25, D.C., at prepaid cost of 15 cents.

† Addendum to National Bureau of Standards Handbook 59, obtainable on request from Publications Section, National Bureau of Standards, Washington 25, D.C.; or see *Radiology* **71,** 263 (1958).

received. Separate sections deal with analysis of extreme cases both with and without opening of the body for surgery or autopsy, accident or injury during surgery or autopsy, contaminated clothing or instruments, cremation, and method of dosage calculation.

Addendum to Report on Protection Against Radiations from Radium, Cobalt-60, and Cesium-137 (National Bureau of Standards Handbook 54).* A complete revision of Handbook 54 is now in preparation. Until this is completed and in order to bring the present handbook into conformance with the new maximum permissible radiation levels,† a two-page addendum to Handbook 54 has been issued. The changes relate to the new permissible levels, the design of teletherapy apparatus, and radiation hazards resulting from fire.

Consideration of exposures under emergency conditions, Subcommittee 14, "Permissible Exposure Dose Under Emergency Conditions" has been working for several years in an endeavor to develop an adequate picture of the effect of large doses of radiation that may be received by large groups of people under emergency or disaster conditions (such as nuclear warfare or the disruption of a large reactor). Doses being considered by this subcommittee are not being regarded as permissible. The report, however, is intended to outline the effects to be expected from large exposures. In the event of nuclear warfare the prime consideration of the population will be immediate survival and therefore large doses may have to be accepted without regard to long-range effects. It is hoped that the handbook when completed will provide some guidance for the policy decisions that will have to be made under emergency conditions.

The program of this committee was started under the acting chairmanship of L. S. TAYLOR. The permanent chairman is now Dr. GEORGE LEROY, Billings Hospital, University of Chicago, Chicago, Illinois.

Call for Papers for Presentation at the Annual Meeting of the Health Physics Society

THE Fourth Annual Meeting of the Health Physics Society will be held from June 18 to 20, 1959 in Galtinburg, Tennessee. Registration will begin on Wednesday afternoon, June 17, 1959.

The Program Committee is accepting abstracts of papers proposed for presentation at this meeting. In order to give an opportunity for somewhat detailed presentations while keeping the total number of papers large, the Committee is planning to divide submitted papers into two groups for simultaneous presentation. In one group the time of each paper will be 10 min, and, after each paper, 5 min will be allotted for discussion. In the other group the time per paper will be 25 min, followed by 5 min of discussion. The tightness of this schedule will necessitate strict adherence to these time limits. The sessions will be run simultaneously in the same building. Arrangements will be made for posting in one session an identification of the paper being presented concurrently in the other. The program will include also an invited paper and two panels of invited speakers. There will be an entertainment program for the wives and children and one afternoon will be reserved for tours and mountain climbing.

Please specify whether your paper is to be considered for a 10 min or a 25 min presentation.

Abstracts of proposed papers should not exceed 250 words in length and should be submitted to the chairman of the 1959 Program Committee: Dr. G. VICTOR BEARD, Health Protection Branch, Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington 25, D.C.

Health Physics Courses Offered by the U.S. Public Health Service's Robert A. Taft Sanitary Engineering Center at Cincinnati, Ohio

Basic Radiological Health

THIS course was offered from March 2 to 13, to provide basic technical knowledge necessary for radiological health work. It was a prerequisite for the courses, "Radiation Protection", "Determination of Radionuclides", and "Environmental Health Aspects of Nuclear Reactor Operations", and was a recommended prerequisite for "Radioactive Pollutants in Air or Water". Basic nuclear physics and chemistry were emphasized. About 50% of the course was devoted to laboratory and problem sessions.

Course items included: (1) sources of radiation exposure; (2) atomic structure and radioactivity: (3) radioactive decay; (4) interaction of radiation with matter; (5) biological effects; (6) maximum permissible levels: (7) principles of radiation detection and protection; (8) survey and personnel monitoring instruments; (9) laboratory counting instruments; (10) assay of beta and gamma emitters; (11) nuclear reactions including fission and fusion; (12) control of radioactive materials.

* Holders of Handbook 54 can secure copies of this addendum by writing to Publications Section, National Bureau of Standards, Washington 25, D.C.

† See footnote on p. 451.

Environmental Health Aspects of Nuclear Reactor Operations

This course offered from March 16 to 25 introduced public health personnel to the overall environmental health and safety aspects of nuclear reactor operations, especially offsite problems. "Basic Radiological Health" was a prerequisite course. Purpose of the course was to provide knowledge of the fundamentals of reactor operations, the reprocessing of spent fuels, disposition of wastes, methods for evaluation of associated hazards, and techniques for establishment of environmental monitoring programs.

The first week of the course, presented at the Sanitary Engineering Center, was followed by a 3-day tour of reactor and waste disposal facilities at the Oak Ridge National Laboratory, with lectures by ORNL staff members.

The course included: (1) principles of nuclear reactors; (2) reactor theory and control; (3) reactor shielding; (4) neutrons—biological effects, detection, and measurement; (5) reactor safety and hazard evaluation; (6) site selection; (7) fuel reprocessing methods; (8) characteristics and disposal of fission products and coolant wastes; (9) environmental surveillance.

Dr. NEWTON UNDERWOOD, Professor of Physics, N. C. State College, Raleigh, N. C., and Mr HAROLD LAMONDS, Director, Raleigh Research Reactor, N. C. State College, Raleigh, N. C., participated as guest lecturers in this course. The Raleigh reactor was the first privately owned unit to be put into operation in the United States.

PAPERS TO BE PUBLISHED IN FUTURE ISSUES

W. H. LANGHAM and E. C. ANDERSON: Cs^{137} biospheric contamination from nuclear weapons tests.

J. D. MCLENDON: Accidental radiation excursion at the Oak Ridge Y-12 Plant—II. Health physics aspects of the accident.

B. KAWIN: Retention and localization of phosphorus-32 in rat ovary.

E. J. STORY and A. W. WALTNER: Activity in building air filters.

E. B. WAGNER and G. S. HURST: Gamma response and energy losses in the absolute fast neutron dosimeter.

A. R. SCHULERT, E. J. HODGES, E. S. LENHOFF and J. L. KULP: Strontium-90 distribution in the human skeleton.

R. H. RITCHIE: Calculations of energy loss under the bias in fast neutron dosimetry.

H. W. PATTERSON, W. N. HESS, B. J. MOYER and R. W. WALLACE: The flux and spectrum of cosmic-ray produced neutrons as a function of altitude.

ERRATUM

K. E. COWSER and F. L. PARKER, *Health Physics* 1, 152–163 (1958).

The legend to Fig. 1 (on p. 154) in this article should read:

FIG. 1. Surface topography and water table elevations of the Oak Ridge National Laboratory area; after H. J. KLEPSER and G. D. DE BUCHANANNE, from ORO-58.

END OF VOL.